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6.0 COMPLIANCE WITH THE RADIOLOGICAL CRITERIA FOR LICENSE TERMINATION

6.1 Site Release Criteria

The site release criteria for the Rancho Seco site will correspond to the radiological criteria for unrestricted use given in 10 CFR 20.1402, or:

- Dose Criterion: The residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem/year, including that from groundwater sources; and

- ALARA Criterion: The residual radioactivity has been reduced to levels that are ALARA.

6.2 Site Conditions

6.2.1 General Description

The site is approximately 25 miles southeast of Sacramento and 26 miles northeast of Stockton in the central valley of California between the foothills of the Sierra Nevada Mountains to the east and the Pacific Coast range bordering the Pacific Ocean to the west.

The plant site's rolling terrain is not directly intersected by any streams; however, drainage from higher levels is well defined and intercepts with runoff streams at lower levels. The plant's grade level of approximately 165 feet above mean sea level (MSL) allows excellent drainage without danger of flooding. The elevation of the site acreage varies from 130 feet to 280 feet above MSL and drainage along natural gullies varies from two to six percent. Runoff from the site drains into a seasonal "No – name" creek that is a tributary to Clay Creek. Clay Creek empties into Hadselville Creek. Hadselville Creek then empties in turn into: Laguna Creek south, Cosumnes River, Mokelomne River, Sacramento River, into the Pacific Ocean via the Sacramento River Delta.

The Rancho Seco site consists of an approximately 87-acre fence-enclosed Industrial Area containing the nuclear facility surrounded by District-owned and District-controlled property totaling 2,480 acres. The District has constructed a 30-acre natural gas-fired power plant on the Rancho Seco site, approximately a half mile south of the Industrial Area boundary. Also within the 2,480 acre site are the 560 acre Rancho Seco Reservoir and Recreation Area; a 50 acre solar power (photo-voltaic) electrical generating station; and the 10 acre, 10 CFR Part 72 licensed Independent Spent Fuel Storage Installation (ISFSI).

A detailed description of applicable Rancho Seco environmental conditions and parameters is provided in Chapter 8, "Supplement to the Environmental Report." These parameters include site and surrounding area physical descriptions including population, topography, vegetation, soil types, surface water quality, climate and meteorology, hydrology and geology. Additionally Chapter 2, Site Characterization, (Section 2.2) contains a detailed description of the Rancho Seco site Hydrogeological Assessment results completed in 2005. The information contained in Chapters 2 and 8 form the basis for determining many of the site-specific dose modeling inputs (see Section 6.6.2).

6.2.2 Remaining Structures at Time of License Termination

In general, especially for structures formerly containing radioactive materials, the structures remaining within the Industrial Area of the site at the time of license termination will be concrete or brick-and-mortar structures, including the cooling towers, with most systems and components removed. The Interim Onsite Storage Building (IOSB) will also remain onsite but will be retained under the 10 CFR 50 license until ultimate disposal of the Class B and Class C radioactive waste that will be stored there.

6.3 Source Term Assumptions

6.3.1 Potential Radionuclides of Concern

As part of the source-term abstraction process, an analysis was performed in Rancho Seco Decommissioning Technical Basis Document DTBD-04-001, "Radionuclides for Consideration During Rancho Seco Nuclear Generating Station Characterization or Final Status Surveys," [Reference 6-1] to identify a suite of radionuclides that could potentially be present on remaining site structural surfaces, in site soils and in groundwater following completion of decommissioning activities. Development of the suite of radionuclides began first with NUREG/CR–3474, "Long-Lived Activation Products in Reactor Materials," [Reference 6-2]. This NUREG assessed the problems posed to reactor decommissioning by long-lived activation products in reactor construction materials. Samples of stainless steel, vessel steel, concrete and concrete ingredients were analyzed for up to 52 elements in order to develop a database of activatable major, minor and trace elements. The suite of radionuclides was developed by combining those radionuclides listed in NUREG/CR–3474 Table 5.6, "Activation of PWR Bioshield (Ci/gm) Average Rebar 30 EFPY at Core Axial Midplane," Table 5.13, "Activity Inventory of PWR Internals at Shutdown (Total Ci)," and Table 5.15, "Inventories of PWR and BWR Vessel Walls at Shutdown (Total Ci)." Only radionuclides with half-lives of two or more years were included on the suite. Radionuclides with half-lives less than two years would not be expected to still be observed since seven or more half-lives would have occurred since final shutdown of the Rancho Seco reactor.

Second, radionuclides with half-lives of two or more years identified in NUREG/CR–4289, "Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants," [Reference 6-3] as being present in PWRs were compared with the suite generated above. NUREG/CR-4289 investigated residual radionuclide concentrations, distributions and inventories at seven nuclear power plants (four shutdown and three operating, including Rancho Seco) to provide a database for use in formulating policies, strategies and guidelines for the eventual decommissioning of retired nuclear power plants. This study addressed radionuclides (both activation and fission products) transported from the reactor pressure vessel and deposited in all other contaminated systems of each nuclear plant. Emphasis was placed on measuring the long-lived radionuclides that are of special concern from a low-level waste management standpoint. The study resulting in NUREG/CR–4289 was a companion study to the study that resulted in NUREG/CR–3474. Any radionuclides identified in NUREG/CR–4289 but not in NUREG/CR–3474, were added to the above suite.

Third, radionuclides with half-lives of two or more years identified in NUREG/CR–0130, "Technology, Safety and Cost of Decommissioning," [Reference 6-4] as being present in PWRs were compared with the list generated above. These radionuclides were identified in NUREG/CR-0130 Table 7.3-9, "Reactor Coolant Radionuclide Concentrations (12) in an

Operating PWR," Table 7.3-10, "Radioactive Surface Contamination in the Reference PWR Resulting from Accumulated Coolant Leakage in an Ion Exchanger Vault (Fractional Activity Normalized at Reactor Shutdown)," and Table 7.3-11, "Isotopic Composition of Accumulated Radioactive Surface Contamination in the Reference PWR (Renormalized for Each Decay Time)." Any radionuclides identified in NUREG–0130 but not in either NUREG/CR–3474 or NUREG/CR-4289, were added to the above suite.

Finally, an ORIGEN computer code run was used to determine if there were additional radionuclides that should be added to the above suite. The ORIGEN code run was based on Cycle 4 through 7 irradiation of selected batch 6 fuel assemblies with a decay period of 13.64 years from shutdown. This resulted in the addition of Pm-147, Pu-241, Am-243 and Cm-243 to the suite.

6.3.2 Discounting Insignificant Radionuclides

Since the suite of radionuclides developed in Section 6.3.1 includes trace elements that would not likely be found at Rancho Seco due to their low abundance, an evaluation of activation product radionuclides that may be discounted from being of potential importance was performed. The inventory for each radionuclide was determined from activity inventories provided in Table 5.13 and Table 5.15 of NUREG/CR-3474. From this information, the percentage of total inventory for each radionuclide was calculated.

The ORIGEN computer code run also contains trace radionuclides that would not likely be found at Rancho Seco due to their low abundance. The radionuclide inventory was determined from the run as well as relative contribution from each radionuclide.

Based on the above evaluations, it was determined that individual radionuclides which contributed less than 0.1 percent of the total activity could be discounted from the suite of identified radionuclides providing that potential dose contributed by the sum of the radionuclides discounted does not exceed one percent of the total calculated dose.

The radionuclides that meet the criterion of contributing less than 0.1 percent of the total activity include:

Cl-36	Ar-39	Ca-41	Mn-53	Se-79	Kr-81
Kr-85	Zr-93	Mo-93	Sn-121m	I-129	Ba-133
Cs-135	Pm-145	Sm-146	Sm-151	Tb-158	Ho-166m
Hf-178m	Pb-205	U-233	Am-243	Cm-243	

Several additional radionuclides meet the criteria of contributing less than 0.1 percent of the total activity but cannot be discounted because they have other methods of production in addition to activation of reactor components and have been observed in 10 CFR Part 61 waste stream analyses or in site characterization samples. These radionuclides include H-3, C-14, Nb-94, Ag-108m, Eu-152, and Pu-239.

In order to evaluate compliance with the dose criteria for discounted radionuclides, the Nuclear Regulatory Commission (NRC) developed computer code DandD, Version 2.1.0, was used to calculate doses for both residential and occupancy scenarios. The DandD code was used with the NRC determined default parameters to represent a conservative screening tool. Input concentrations for each radionuclide used in the residential scenario were their percent of total activity input as concentration in pCi/g. Input concentrations for each radionuclide used in the occupancy scenario were 1,000 times their percent of total activity input as surface contamination in dpm/100 cm^2. DandD does not support the following radionuclides and could not calculate their dose contribution:

Ar-39	Mn-53	Kr-81	Kr-85	Ag-108m	Ba-133
Pm-145	Sm-146	Tb-158	Hf-178m	Pb-205	

Therefore doses could be calculated for only the following discounted radionuclides:

Cl-36	Ca-41	Se-79	Zr-93	Mo-93	Sn-121m
I-129	Cs-135	Sm-151	Ho-166m	U-233	Am-243
Cm-243					

The calculated dose from discounted NUREG radionuclides represents only 3.73E-02 percent and dose from discounted ORIGEN radionuclides represents only 4.27E-02 percent of the total calculated dose for the residential scenario. The calculated dose from discounted NUREG radionuclides represents only 1.99E-03 percent and dose from discounted ORIGEN radionuclides represents only 5.53E-01 percent for the of the total calculated dose occupancy scenario. Therefore, it is appropriate to discount these radionuclides.

The activity represented by the radionuclides not supported by the DandD code is calculated to be only 4.23E-03 percent of the total activity presented in NUREG/CR-3474. Of these radionuclides, Ar-39, Kr-81 and Kr-85 are noble gases and it is highly unlikely that they would still be present in soil and on structural surfaces. Therefore, it is appropriate to discount Ar-39, Kr-81 and Kr-85.

Potential dose contribution from the remaining radionuclides not supported by the DandD code was evaluated by comparison of the inhalation and ingestion exposure-to-dose conversion factors (DCFs) contained in Federal Guidance Report No.11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," [Reference 6-5]. Weighted DCFs were calculated for each discounted radionuclide and summed for both inhalation and ingestion DCFs. These totals were then compared to the sum of the weighted DCFs for the two most abundant radionuclides, Co-60 and Ni-63. This resulted in a total of 5.36E-03 percent for inhalation DCFs and 1.25E-03 percent for ingestion DCFs. Therefore, it is appropriate to discount all of the radionuclides not supported by the DandD code.

Although included in the suite of theoretical radionuclides, the naturally occurring radionuclides K-40, U-234, U-235, U-236 and U-238 were not detected in characterization survey samples at concentrations distinguishable from naturally occurring concentrations. Therefore, these radionuclides have been discounted from any further consideration.

Radioactive waste streams are periodically sampled and analyzed at Rancho Seco. Analyses are performed for radionuclides listed in 10 CFR 61.55 Tables 1 and 2 as well as other supplementary radionuclides on a select basis. The potential radionuclides identified for discounting as described above were compared with the Rancho Seco 2003 Waste Stream Evaluation [Reference 6-6]. None of these radionuclides were identified as being present at Rancho Seco. However, an additional radionuclide, Pu-242, had been identified by waste stream analysis and was added to the suite of radionuclides. The resulting suite of radionuclides is considered to be site-specific to Rancho Seco and considered to be potentially present on remaining site structural surfaces, in site soils and in groundwater following completion of decommissioning activities. This site-specific suite of radionuclides is listed in Table 6-1.

Table 6-1

Site-Specific Suite of Radionuclides for Use at Rancho Seco

Radionuclide	Half Life (Years)	Decay Mode	Radionuclide	Half Life (Years)	Decay Mode
H-3	1.23E+01	β	Cs-137	3.02E+01	β
C-14	5.73E+03	β	Pm-147	2.62E+00	β
Na-22	2.60E+00	β^+, γ	Eu-152	1.36E+01	β, γ
Fe-55	2.70E+00	γ	Eu-154	8.80E+00	β, γ
Ni-59	7.50E+04	γ	Eu-155	4.96E+00	β, γ
Co-60	5.27E+00	β, γ	Np-237	2.14E+06	α, γ
Ni-63	1.00E+02	β	Pu-238	8.78E+01	α, γ
Sr-90	2.86E+01	β	Pu-239	2.41E+04	α, γ
Nb-94	2.03E+04	β, γ	Pu-240	6.60E+03	α, γ
Tc-99	2.13E+05	β, γ	Pu-241	1.44E+01	β
Ag-108m	1.27E+02	γ	Am-241	4.32E+02	α, γ
Sb-125	2.77E+00	β, γ	Pu-242	3.76E+05	α, γ
Cs-134	2.06E+00	β, γ	Cm-244	1.81E+01	α, γ

α – Alpha decay
β – Beta decay
β^+ – Positron decay
γ – Gamma decay

6.4 Dose Modeling Considerations

6.4.1 Overview

The NRC states in NUREG–1757, Volume 2, "Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria," [Reference 6-7] that, *"generally, the licensee's dose modeling should have one of the following objectives:*

1. *Develop DCGLs commensurate with demonstrating compliance with the dose-based release criterion, and then demonstrate through FSS that residual radioactivity concentrations at the site are equal to or below the DCGLs.*

2. *Assess dose associated with actual concentrations of residual radioactivity distributed across the site to determine whether the concentrations will result in a dose that is not equal to or below the regulatory dose criterion."*

The District has chosen to use the first of these objectives and will demonstrate at the time of the final status survey (FSS) before release that residual radionuclide concentrations across the site are below a pre-specified concentration limit with a pre-specified degree of confidence. The design of the FSS is based on the proposed derived concentration guideline levels (DCGLs), in accordance with MARSSIM.

The approach taken to dose modeling for the site is consistent with the information provided in Chapter 5 and Appendix I of NUREG- 1757, Volume 2 for site-specific modeling, including the information regarding source term abstraction, scenarios, pathways and critical groups. The dose model is defined by the three factors: 1) the scenario, 2) the critical group and 3) the exposure pathways.

The approach outlined above was used to develop dose models to calculate DCGLs for the following:

- Surface and subsurface soil,
- Buried piping,
- Structural surfaces,
- Embedded piping, and
- Bulk material.

6.4.2 Industrial Worker Scenario for Surface and Subsurface Soil Exposure

6.4.2.1 Industrial Worker Scenario Justification

The District has no plans to release all or part of the District-owned and District-controlled 2,480 acre site for ownership by members of the public. Although the public does have access to a 560 acre Rancho Seco Reservoir and Recreation Area (see Figure 6-1) existing within the 2,480 acre site, the public does not have ready access to the remaining areas of the site. The Reservoir and Recreation Area are located in a non-impacted area hydrogeologically upgradient from the impacted Industrial Area. The reservoir water is replenished by water from the Folsom-South canal, constructed by the U.S. Bureau of Reclamation. No surface water runoff from the Industrial Area or the discharge canal or impacted depression area soils west of the Industrial Area enters the reservoir.

The entire site is owned by the District and is not a tax burden on the District because the District is a tax-exempt governmental organization. Thus there is no enticement to sell portions of the site because it is a tax burden to the District.

The site continues to be an important electrical distribution center for the District. The Rancho Seco switchyard (see Figure 6-2) has been in continual use since permanent shutdown of Rancho Seco and is a major intertie connecting the District's electrical transmission and distribution system with the Western Grid transmission system. There are six energized 220/230-kV transmission lines leaving the switchyard, four exiting the site to the west and two to the south.

A 50-acre photovoltaic (PV) generating facility is located on a non-impacted portion of the 2,480-acre site (see Figure 6-3). This generating facility using solar power consists of six solar arrays (PV1 – PV6) with a total generating capacity of 3.2 MWe. The first phase of the facility became operational in August 1984. This photovoltaic generating facility is connected to the District's electrical transmission and distribution system.

A new 500 MWe natural gas fueled combined-cycle generating facility is located on a 30-acre non-impacted portion of the 2,480-acre site and began commercial operation in February 2006 (see Figure 6-4). This $435 million natural gas fueled generating facility is connected to the District's electrical transmission and distribution system through the Rancho Seco switchyard.

The District operates one of thirty-four Control Areas in the Western Interconnection of the North American Electric Reliability Council (NERC). The Western Interconnection includes the provinces of Alberta and British Columbia, the northern portion of Baja California, Mexico, and all or portions of the 14 western states in between. The District Control Area is responsible for the continuous reliable operation of the transmission system that is owned by: the District; the Western Area Power Administration; and the participants of the California-Oregon Transmission Project, the cities of Redding, Shasta, Roseville, and Modesto, California. In addition, the District control area includes the customer load and generation that are connected to this transmission system. The District Control Area, in Northern California, extends from the California-Oregon border to Modesto, California.

NERC requires the District Control Area to continue to operate the transmission system reliably without relying on any of the elements in the primary control center. To meet this requirement, the District is in the process of building a $13.3 million Backup Control Center (BCC) in the Administration Building at Rancho Seco. The District intends to indefinitely maintain BCC operations at the Rancho Seco site.

The Rancho Seco concrete structures will remain in place after equipment removal and any required decontamination/remediation. There are no current plans for reutilization of these structures. Most easily demolished structures will be removed prior to license termination.

For the reasons stated above, it is reasonable to assume that the District will retain ownership of the site for the foreseeable future and that members of the public will not have ready access to impacted areas of the site.

6.4.2.2 Critical Group for Surface and Subsurface Soil Exposure

The average member of the critical group is defined as a District employee or contractor who is allowed occupational access to impacted areas of the site over the course of his/her employment. The assumption is made that occupancy would be limited to a 50-workweek year (2,000 hours per year). It was further assumed that the industrial worker would spend half of his/her time indoors and half outdoors while onsite.

6.4.2.3 Site-Specific Exposure Pathways

Under the industrial worker scenario the average member of the critical group receives potential exposure from contaminated soil by direct exposure, inhalation of contaminated soil that becomes airborne and ingestion of contaminated soil. The industrial worker could also receive potential exposure from drinking water or buried piping.

As discussed in Section 6.5.1, RESRAD was chosen as the computational method to calculate soil DCGLs. The industrial worker scenario varies significantly from the residential farmer scenario by allowing the following less conservative but realistic assumptions. These less conservative assumptions are realistic based upon the scenario justifications discussed in Section 6.4.2.1:

- Suppression of the plant ingestion pathway,
- Suppression of the meat ingestion pathway,
- Suppression of the aquatic foods pathway, and
- Drinking water pathway is not suppressed – there are currently four potable water wells existing on the 2,480-acre site. Three of these wells are up gradient of the impacted area; however, the fourth well is in the northern portion of the impacted area.

6.4.3 Industrial Worker Scenario for Building Occupancy Exposure

6.4.3.1 Industrial Worker Scenario Justification

The justification provided in Section 6.4.2.1 for using an industrial worker scenario for evaluating exposure to contaminated surface and subsurface soils also applies to using an industrial worker scenario for building occupancy and exposure to contaminated surfaces, bulk materials and embedded piping.

6.4.3.2 Critical Group for Structural Surface Exposure

The average member of the critical group is defined as a District employee or contractor who is allowed occupational access to impacted areas of the site over the course of his/her employment. The occupancy assumed is the 45 hours per week used in NUREG/CR-5512, Volume 3, "Residual Radioactive Contamination from Decommissioning – Parameter Analysis," [Reference 6-8].

6.4.3.3 Site-Specific Exposure Pathways

As discussed in Section 6.5.2, RESRAD-BUILD was chosen as the computational method to calculate structural surface DCGLs. The RESRAD-BUILD code is a pathway analysis model designed to evaluate the potential radiological dose to an individual who works in a building contaminated with radioactive material. It considers the releases of radionuclides into the indoor air by diffusion, mechanical removal, or erosion. RESRAD-BUILD considers seven exposure pathways:

1. External exposure directly from the source,
2. External exposure to materials deposited on the floor,
3. External exposure due to air submersion,
4. Inhalation of airborne radioactive particulates,
5. Inhalation of aerosol indoor radon progeny (in the case of the presence of radon predecessors) and tritiated water vapor (the radon pathway was turned off because the nuclear regulatory commission (NRC) does not regulate dose received from radon and progeny),
6. Inadvertent ingestion of radioactive material directly from the source, and

7. Ingestion of materials deposited on the surfaces of the building compartments.

6.5 Computational Model Used for Dose Calculations

6.5.1 Impacted Area Soils

The computer code RESidual RADioactive materials (RESRAD) v6.22, followed by v6.3 after its release during the summer of 2005, was selected to perform site-specific dose modeling of impacted area soils because of the ability to model subsurface soil contamination contained within the code. Argonne National Laboratory (ANL) developed the RESRAD computer code under the sponsorship of the U.S. Department of Energy (DOE). The code has been used widely by DOE and its contractors, the U.S. NRC, U.S. Environmental Protection Agency (EPA), U.S. Army Corps of Engineers, industrial firms, universities, and foreign government agencies and institutions. This code is a pathway analysis model designed to evaluate potential radiological doses to an average member of the specific critical group.

The NRC has adopted a risk-informed approach in assessing impacts on the health and safety of the public from radioactive contamination remaining at decommissioned sites. Therefore, the NRC tasked ANL to develop parameter distribution functions and parametric analysis for RESRAD for conducting probabilistic dose analysis. As part of this effort, external modules equipped with probabilistic sampling and analytical capabilities were developed for the RESRAD code. The modules are also equipped with user-friendly input/output interface features to accommodate numerous parameter distribution functions and to fulfill results display requirements.

The RESRAD database includes inhalation and ingestion dose conversion factors from the EPA's Federal Guidance Report (FGR) No. 11, direct external exposure dose conversion factors from FGR-12 and radionuclide half-lives from International Commission on Radiological Protection Publication 38 [References 6-9, 6-10 and 6-11 respectively].

6.5.2 Impacted Structural Surfaces and Bulk Material

RESRAD-BUILD v3.22, followed by v3.3 after its release during the summer of 2005, was selected to perform site-specific dose modeling of impacted structural surfaces and bulk material. RESRAD-BUILD is a computer code designed to evaluate the radiation doses from RESidual RADioactivity in BUILDings. The RESRAD-BUILD code was developed by ANL under sponsorship of the U.S. DOE and other federal agencies.

The RESRAD-BUILD computer code is a pathway analysis model designed to evaluate the potential radiological dose incurred by an individual who works or lives in a building contaminated with radioactive material. The transport of radioactive material within the building from one compartment to another is calculated with an indoor air quality model. The air quality model considers the transport of radioactive dust particulates and radon progeny due to air exchange, deposition and resuspension, and radioactive decay and ingrowth.

Seven exposure pathways are considered in the RESRAD-BUILD code: (1) external exposure directly from the source, (2) external exposure from materials deposited on the floor, (3) external exposure due to air submersion, (4) inhalation of airborne radioactive particulates, (5) inhalation of aerosol indoor radon progeny and tritiated water vapor, (6) inadvertent ingestion of radioactive material directly from the source, and (7) ingestion of materials deposited on the surfaces of the building compartments. Various exposure scenarios may be modeled with the RESRAD-BUILD code. These include, but are not limited to, office worker, renovation worker, decontamination worker, building visitor, and residency scenarios. Both deterministic and probabilistic dose analyses can be performed with RESRAD-BUILD, and the results can be shown in both text and graphic reports.

6.5.3 Buried Piping

The buried piping scenario utilizes soil DCGL values derived in Section 6.6.2. Under the scenario, buried piping is assumed to disintegrate instantaneously upon license termination. The disintegrated media is assumed to be soil and the media volume is assumed to be equal to the piping volume. A gross DCGL value applicable to interior piping surfaces was derived using standard computational methods assuming the disintegrated media is contaminated to soil DCGL concentrations using average observed nuclide fractions for soil and piping surface contamination.

Potential dose to the receptor at one meter above the surface soil was evaluated assuming a soil cover depth of 0.305 meter and 1.0 meter. The latter depth is considered a nominal depth for buried piping that will remain on site after license termination. The MicroShield® computer code was used to perform these calculations. MicroShield® is a comprehensive photon/gamma ray shielding and dose assessment program.

6.5.4 Embedded Piping

The embedded piping scenario assumes that the piping remains in place following decommissioning and that the dose to the industrial worker is from direct gamma exposure from the residual activity in the pipe with allowance made for photon attenuation by the wall or floor thickness of concrete remaining over the pipe. Whole body dose from the embedded pipe is considered additive along with the dose to the industrial worker resulting from residual activity on the walls or floors of the room or area in which the embedded pipe is present. The surface DCGL will be reduced as necessary by the dose contribution from the embedded piping in order to ensure compliance with the annual dose limit.

The MicroShield® computer code was used to evaluate dose from embedded piping. MicroShield® is a comprehensive photon/gamma ray shielding and dose assessment program.

6.6 <u>Derived Concentration Guideline Levels (DCGLs)</u>

6.6.1 RESRAD/RESRAD-BUILD Parameter Treatment

6.6.1.1 Parameter Classification

RESRAD and RESRAD-BUILD parameters are classified as behavioral, metabolic or physical. Some parameters may belong to more than one of these types.

In 1999, the NRC tasked Argonne National Laboratory with adapting the existing RESRAD code for use in site-specific dose modeling and analysis in accordance with the NRC's guidance in the Standard Review Plan (SRP) for Decommissioning to demonstrate compliance with the license termination rule. For this reason, ANL revised and customized the code to be consistent with the current NRC guidance for both deterministic and probabilistic dose modeling being developed in the SRP for Decommissioning. The first step in the procedure used by Argonne to develop the probabilistic code involved listing and classifying the parameters. The parameter classification has been documented in Attachment A to NUREG/CR-6697, "Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes," [Reference 6-12].

Physical parameters are determined by the source, its location, and geological characteristics of the site (i.e., these parameters are source- and site-specific). These include the geohydrological, geochemical, and meteorological characteristics of the site. The characteristics of atmospheric and biospheric transport up to, but not including, uptake by, or exposure of, the dose receptor, would also be considered physical input parameters.

A behavioral parameter is any parameter whose value would depend on the receptor's behavior and the scenario definition. For the same group of receptors, a parameter value could change if the scenario changed (e.g., parameters for recreational use could be different from those for residential use).

If a parameter represents the metabolic characteristics of the potential receptor and is independent of scenario, it is classified as a metabolic parameter. The parameter values may be different in different population age groups. According to the recommendations of the International Commission on Radiological Protection, Report 43 (ICRP 1985) [Reference 6-13], parameters representing metabolic characteristics are defined by average values for the general population. These values are not expected to be modified for a site-specific analysis because the parameter values would not depend on site conditions.

ANL ranked physical parameters by priority as 1, 2, or 3 where 1 represents high priority, 2 represents medium priority and 3 represents low priority. This ranking was the second step in the procedure used by ANL to develop the probabilistic code. The parameter ranking has been documented in Attachment B to NUREG/CR-6697.

6.6.1.2 Parameter Treatment

The parameters were treated as deterministic or stochastic depending on parameter type, priority, and availability of site-specific data and the relevance of the parameter in dose calculations. Deterministic modules of the code use single values for input parameters and generate a single value for dose. Probabilistic versions of the code use probability distributions for input parameters and generate a range of doses. Stochastic parameters are parameters that are defined by a probabilistic distribution.

The behavioral and metabolic parameters were treated as deterministic. The physical parameters for which site-specific data were available were also treated as deterministic. The remaining physical parameters for which no site-specific data were available to quantify were classified as either priority 1, 2 or 3. Priority 1 and 2 parameters were treated as stochastic. The priority 3 physical parameters were treated as deterministic.

6.6.2 Surface and Subsurface Soils

6.6.2.1 Mathematical Hydrogeological Model

RESRAD requires that the hydrogeological conditions of the site be described in a simplified mathematical model from the surface down to the first saturated potable groundwater zone.

RESRAD allows for the modeling of an uncontaminated cover, a contaminated soil zone, up to five unsaturated (vadose) zones and a saturated zone. The Rancho Seco hydrogeological information used to develop the simplified mathematical model is based on the original siting study performed in 1967 and 1968 and described in the Rancho Seco Final Safety Analysis Report (FSAR).

The subsurface exploration performed for the original siting study included:

1. The excavation of 22 backhoe trenches where natural outcrops were scarce;

2. The drilling and logging of twenty-eight 24-inch bucket auger holes, totaling 1,552 feet (generally, these auger holes were drilled to 70 feet unless special considerations dictated otherwise);

3. Twenty-two small-hole borings (3 ⅞ to 4 ¾ inch), totaling 2,016 feet, for sampling and logging;

4. A core hole, 602 feet deep, which was visually and geophysically logged, and which provided 23 representative core samples for laboratory testing of unconfined compression, absorption, porosity, and apparent and bulk specific gravity (this hole was used in testing for possible yield of ground water and a piezometer was installed for periodic measuring of depth to the ground water surface); and

5. A shallow seismic refraction survey, covering three lines and totaling 11,550 lineal feet, to determine seismic velocities of the foundation material and depths to significant velocity layers.

Boring 104 is closest to the location of the major plant structures. The undisturbed surface elevation of this boring was +174.2 feet msl. In this boring, silty gravel was encountered to a depth of approximately 10 feet (elevation +164), underlain by poorly graded sand to a depth of approximately 20 feet (elevation +154). Below 20 feet, sandstone and siltstone were encountered to a depth of approximately 105 feet (elevation +70). At 105 feet, sandstone was encountered to termination of the boring at 115 feet (elevation (elevation +60).

Boring 301 was drilled at the approximate center of the reactor containment building. Silty, sandy gravel was encountered to a depth of 10 feet and alternating sandstone and siltstone layers to a depth of approximately 73 feet. Sandstone was encountered from 73 feet to 137 feet; alternating sandstone and siltstone were again encountered to a depth of 366.5 feet. A green tuff was logged at 366.5 feet to the bottom of the boring, which terminated at 390 feet. Geologically, the boring penetrated the Laguna formation and the Mehrten formation, and was bottomed in the Valley Springs formation.

A 6¼ inch hole was drilled from a surface elevation of +177.4 feet to a depth of 602 feet (elevation –425) to obtain geologic and seismic data on the deep foundation materials. Groundwater was encountered in this hole at an elevation of +34.5 feet.

The current plant configuration has established the plant finished grade at elevation +165 feet which required excavation of 10 to 20 feet of material at the site. The foundation for the reactor containment building has been established at approximately 35 feet below the plant finished grade.

Based upon the above observations, the following simplified mathematical model is proposed for RESRAD calculations:

- A contaminated silt layer 15 cm thick,
- An unsaturated silt layer 1 foot (0.305 meters) thick,
- An unsaturated fine sand layer 10 feet (3.05 meters) thick,
- An unsaturated siltstone layer 84 feet (25.6 meters) thick,
- An unsaturated sandstone layer 35.5 feet (10.8 meters) thick, and
- An underlying saturated sandstone layer.

6.6.2.2 Radionuclides of Concern

Derivation of a site-specific suite of radionuclides potentially present at Rancho Seco was discussed in Section 6.3. This suite of potential radionuclides contains a total of 26 radionuclides. On May 28, 2004, Rancho Seco submitted a spent fuel pool cooler pad soil sample collected on March 8, 2004, to General Engineering Laboratories (GEL) for analysis of the entire suite of 26 radionuclides potentially present. This sample was known by onsite gamma spectroscopy to have the highest level of contamination of any soil samples collected during the site characterization process. Of the suite of 26 potential radionuclides, GEL positively identified only six radionuclides. These were C-14, Co-60, Ni-63, Sr-90, Cs-134, and Cs-137. Potassium-40 and U-238 were detected but they are naturally occurring radionuclides and the levels at which they were detected are indistinguishable from background levels. Therefore, K-40 and U-238 were discounted from further analysis. Single nuclide DCGL values were derived for each of the six radionuclides detected by GEL.

6.6.2.3 Sensitivity Analysis of Detectable Radionuclides

A sensitivity analysis was performed first to identify the parameters that are sensitive in the industrial worker scenario for the detectable radionuclides. As shown schematically in Figure 6-5, the parameter selection process starts with the selection of the RESRAD-BUILD parameter to be evaluated. If the parameters were classified as physical, then they were reviewed to determine if measured, site-specific values or look-up values based on soil type for the parameters were available. If measured, look-up or site-specific values for physical parameters were not available; the parameters were then ranked by priority as 1, 2, or 3 where 1 represents high priority, 2 represents medium priority and 3 represents low priority. This ranking was the second step in the procedure used by ANL to develop the probabilistic code. If the physical parameters were ranked as priority 3, the assigned default values in RESRAD v6.22 were used for performing sensitivity analyses. ANL developed statistical parameter distributions for the physical parameters ranked as priority 1 or 2. These parameter distributions have been documented in Attachment C to NUREG/CR-6697. The parameters selected for sensitivity analysis and their selection justification are provided in Appendix 6-A and the statistical parameter distributions used are provided in Appendix 6-B.

The first sensitivity analysis was performed using uncorrelated parameters. The site-specific RESRAD v6.22 dose model was loaded with the simplified mathematical model parameters contained in Appendix 6-A then with the statistical parameter distributions provided in

Appendix 6-B and run in the probabilistic mode. The uncertainty analysis input settings for this calculation were:

- Latin Hypercube sampling,
- Random seed – 1000,
- Number of observations – 300,
- Number of repetitions – 1, and
- Grouping of observations – correlated or uncorrelated.

Radionuclide concentrations used for the calculation were values obtained by GEL for sample number SC 8100010 DS08A1 (spent fuel cooler pad soil). This was a single soil sample that represented the most highly contaminated soil sampled onsite as indicated by onsite gamma isotopic analysis. The sample results obtained by GEL were decayed from the date of analysis to July 1, 2008 to represent the radionuclide mixture at the approximate completion of final status surveys. The RESRAD calculation was performed using the entire suite of identified radionuclides with their concentrations expected to exist on July 1, 2008. This approach identifies the sensitive parameters for the entire mixture, not just individual radionuclide parameter sensitivity. Radionuclide concentrations are provided in Table 6-2 below.

Table 6-2

Sensitivity Analysis Radionuclide Concentrations

Radionuclide	Analysis Date	Analysis Result (pCi/g)	Half Life (Years)	Decayed Conc. (pCi/g)
C-14	06/24/04	4.74E+00	5.73E+03	4.74E+00
Co-60	06/22/04	1.13E+01	5.27E+00	6.66E+00
Ni-63	06/24/04	1.75E+02	1.00E+02	1.70E+02
Sr-90	07/08/04	1.41E+00	2.86E+01	1.28E+00
Cs-134	06/22/04	8.90E-01	2.06E+00	2.30E-01
Cs-137	06/22/04	1.04E+03	3.02E+01	9.48E+02

The absolute value of the calculated partial ranked correlation coefficient (PRCC) of the peak of the mean dose was then used to classify the parameters with statistical distributions as sensitive or non-sensitive. PRCC was chosen because NUREG/CR-6692, “Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes,” [Reference 6-14] recommends that it be used when nonlinear relationships, widely disparate scales or long tails are present in the inputs and outputs. If the absolute value of the PRCC was greater than 0.25, then the parameter was classified as sensitive. If the absolute value of the PRCC was equal to or less than 0.25, then the parameter was classified as non-sensitive.

Finally, values for use in dose modeling for the physical parameters with sensitive parameters were selected based on sensitivity of the calculated PRCC following the guidance of NUREG/CR-6676, “Probabilistic Dose Analysis Using Parameter Distributions Developed for RESRAD and RESRAD-BUILD Computer,” [Reference 6-15]. If the absolute value of the PRCC was greater than 0.25, then the parameter value at either the 75% quartile or the 25% quartile was selected based on TEDE correlation with the parameter. If the PRCC value of the peak of the mean dose was negative, the parameter to dose correlation is negative and the parameter value at the 25% quartile was selected. If the PRCC value was positive, the parameter to dose correlation is positive and the parameter value at the 75% quartile was selected.

The parameter values were obtained from the RESRAD probabilistic calculation results using the interactive output feature of the uncertainty results. A double click on the left mouse button opens the interactive output dropdown window. From the interactive output dropdown window the “Results” folder is selected. From the “Results” folder the “Graphics” sub-folder is selected. The “Cumulative Density” is then selected as the Plot Type and the “Input Vector” is selected as the Primary Object. The parameter value is determined by a right mouse button click on the plot and selecting “Edit Chart Data” from the dropdown window. This opens the Data Grid Editor dropdown window. From this window, 0.25 or 0.75 is selected, as appropriate from the C2 column, which represents the appropriate quartile value. The corresponding parameter value is contained in the C1 column. The sensitive parameters for each radionuclide and the maximum calculated sensitive parameter PRCC values are listed in Appendix 6-C.

A few input parameters are clearly related, such as effective porosity and total porosity. NUREG/CR-6697, Attachment C identifies potential correlations among RESRAD parameters assigned statistical distributions. These correlations (for which RESRAD v6.22 allows correlating and a statistical distribution was used in the uncorrelated parameter distribution calculation) are provided in Table 6-3 below.

Table 6-3
Potential Parameter Correlations

Parameter	Correlated With	Positive/Negative Correlation
Effective porosity of unsaturated zone 1	Total porosity of unsaturated zone 1	Strong positive
Effective porosity of unsaturated zone 2	Total porosity of unsaturated zone 2	Strong positive
Contaminated zone Erosion rate	Runoff coefficient	Positive
Contaminated zone soil density	Contaminated zone total porosity	Negative
Unsaturated zone 1 soil density	Unsaturated zone 1 total porosity	Negative
Unsaturated zone 2 soil density	Unsaturated zone 2 total porosity	Negative

The calculation with uncorrelated parameters performed above was then repeated with the addition of the potential parameter correlations contained in Table 6-3 as additional parameter inputs for the uncertainty analysis. In the correlated/uncorrelated grouping, the RESRAD v6.22 user specifies the degree of correlation between each correlated parameter by inputting the correlation coefficients between the ranks of the parameters. If the correlation identified in Table 6-3 was indicated to be positive, a rank correlation coefficient of 0.5 was used. If the correlation is indicated to be negative, a rank correlation coefficient of -0.5 was used. If the correlation is indicated to be strong positive, a rank correlation coefficient of 0.9 was used.

The PRCC values for each parameter and the sensitive parameter values assigned are listed in Appendix 6-C. As can be seen from the results provided in Appendix 6-C, there is insignificant difference between PRCC values calculated using uncorrelated or correlated parameters and, in each calculation, the same three sensitive parameters were identified. These parameters include density of the contaminated zone, contaminated zone K_d value for Cs-137 and external gamma shielding factor. The PRCC value for the parameter contaminated zone erosion rate just exceeds the value to classify it as a sensitive parameter for the uncorrelated calculation but is just below the same value for the correlated calculation making it a non-sensitive parameter when correlated. Therefore, it was treated as a non-sensitive parameter when calculating DCGL values because correlated parameter pairs are considered to give a more appropriate indication of individual parameter sensitivity. The sensitive parameters were treated deterministically for calculating DCGL values using the assigned parameter values of Appendix 6-C.

6.6.2.4 Potential Dose from Discounted Radionuclides

The guidance contained in NUREG-1757, Vol. 2, Section 3.3, states "it is reasonable that radionuclides or pathways that are insignificant contributors to dose may be eliminated from further detailed consideration." The guidance further states, "NRC staff considers radionuclides and exposure pathways that contribute no greater than 10 percent of the dose criteria to be insignificant contributors." Because the dose criteria are performance criteria, this 10 percent limit for insignificant contributors is an aggregate limitation only. That is, the sum of the dose contributions from all radionuclides and individual scenario pathways considered insignificant should be no greater than 10 percent of the dose criteria. No limitation on either single radionuclides or pathways is necessary.

However, once it has been demonstrated that radionuclides or exposure pathways are insignificant, then (a) the dose from the insignificant radionuclides and pathways must be accounted for in demonstrating compliance, but (b) the insignificant radionuclides and pathways may be eliminated from further detailed evaluations. Therefore, it is necessary to calculate potential dose from undetected radionuclides and compare the aggregate potential dose to ensure that it does not exceed 10% of the dose limit (i.e., 2.5 mrem/year TEDE).

As discussed in Section 6.6.2.2, only six of the potential radionuclides derived in DTBD-04-001 are considered to be radionuclides of concern for Rancho Seco impacted area surface soil. This leaves 20 discounted radionuclides to be evaluated for potential dose. This potential dose is based on the minimum detectable activity (MDA) value for each hard-to-detect radionuclide analyzed by GEL for the spent fuel pool cooler pad soil sample, decayed, as for the radionuclides of concern, to a FSS date of July 1, 2008. The MDA value for readily detectable radionuclides is based on the average MDA from 17 spent fuel pool cooler pad surface soil samples analyzed by onsite gamma spectroscopy. A decay correction was applied to these average MDA values to correspond to an approximate date of July 1, 2008. The resulting concentrations are provided in Table 6-4.

Table 6-4
Discounted Radionuclide Concentrations for Dose Evaluation

Radionuclide	Analysis Date	Analysis MDA (pCi/g)	Decayed Conc. (pCi/g)
H-3	06/24/04	6.82E+00	5.44E+00
Na-22	***Various***	***3.28E-02***	***9.46E-03***
Fe-55	06/24/04	3.56E+00	1.27E+00
Ni-59	06/22/04	3.58E+00	3.58E+00
Nb-94	***Various***	***3.95E-02***	***3.95E-02***
Tc-99	06/28/04	2.35E+00	2.35E+00
Ag-108m	***Various***	***5.64E-02***	***5.50E-02***
Sb-125	***Various***	***4.03E-01***	***1.25E-01***
Pm-147	06/25/04	2.76E+00	9.55E-01
Eu-152	***Various***	***2.23E-01***	***1.76E-01***
Eu-154	***Various***	***9.04E-02***	***6.26E-02***
Eu-155	***Various***	***3.00E-01***	***1.56E-01***
Np-237	06/22/04	4.41E-01	4.41E-01
Pu-238	06/22/04	8.51E-01	8.24E-01
Pu-239	06/22/04	2.07E-01*	2.07E-01
Pu-240	06/22/04	2.07E-01*	2.07E-01
Pu-241	07/04/04	5.90E+01	4.87E+01
Am-241	06/25/04	7.62E-01	7.57E-01
Pu-242	06/22/04	5.94E-01	5.94E-01
Cm-244	06/25/04	5.60E-01**	4.80E-01

Note: Values in ***bold italics*** are the average MDA from onsite gamma isotopic analysis of 17 spent fuel cooler pad surface soil samples.

*Half of the Pu-239/Pu-240 combined MDA
**Half of the Cm-243/Cm-244 combined MDA

6.6.2.4.1 Identification of Discounted Radionuclide Sensitive Parameters

In order to calculate potential dose from discounted radionuclides, the discounted radionuclides were first evaluated to identify sensitive RESRAD parameters to be treated deterministically in the dose calculation. The site-specific mathematical model developed in Section 6.6.2.3 was used with uncorrelated parameter distributions to perform the sensitivity analysis. The RESRAD v6.22 parameters used are provided in Appendix 6-D and the probabilistic parameter distributions and sensitivity analysis results are provided in Appendix 6-E.

6.6.2.4.2 Calculation of Potential Dose from Discounted Radionuclides

Potential dose from discounted radionuclides was calculated using RESRAD v6.22 in the probabilistic mode using the decayed radionuclide concentrations provided in Table 6-4 and selecting the peak of the mean dose. The site-specific mathematical model used to identify sensitive parameters in Section 6.6.2.4.1 was used with uncorrelated parameter distributions to perform the dose calculations. The RESRAD v6.22 parameters used were taken from

Appendix 6-D and the probabilistic parameter distributions and sensitive parameter value assignments were taken from Appendix 6-E. Sensitive parameters were treated deterministically and non-sensitive parameters were treated stochastically. The total potential dose from all transuranic and non-transuranic discounted radionuclides is 0.572 mrem/year, which occurs during the first year following July 1, 2008.

6.6.2.5 Single Nuclide $DCGL_W$ Values for Detectable Radionuclides

Single nuclide $DCGL_W$ values (soil contamination at the $DCGL_W$ value results in a 25 mrem/year TEDE dose from the single radionuclide) for the detectable radionuclides of concern were calculated by performing an individual RESRAD calculation for each of the six detectable radionuclides. The site-specific RESRAD v6.22 dose model was first loaded with the simplified mathematical model parameters contained in Appendix 6-A (with the exception that a concentration of 1 pCi/g was used for each radionuclide) then with the statistical parameter distributions provided in Appendix 6-B. If a parameter was identified as sensitive in Section 6.6.2.3, it was treated deterministically and the sensitive parameter values listed in Appendix 6-C were used instead of a statistical parameter distribution. RESRAD was then run in the probabilistic mode for each detected radionuclide. The uncertainty analysis input settings for these calculations were:

- Latin Hypercube sampling,
- Random seed – 1000,
- Number of observations – 300,
- Number of repetitions – 1, and
- Grouping of observations – correlated or uncorrelated.

These calculations provided the peak of the mean dose in mrem/year per pCi/g for each detected radionuclide that is listed in Table 6-5. $DCGL_W$ values were then calculated by dividing the regulatory dose limit of 25 mrem/year minus the potential dose contribution from discounted radionuclides of 0.572 mrem/year (or a total of 24.4 mrem/year) by the calculated peak of the mean dose (mrem/year per pCi/g). Results of these calculations are also listed in Table 6-5. Derivation of soil DCGLs is detailed in DTBD-04-005, "DCGLs for Rancho Seco Industrial Area Surface Soils," [Reference 6-16].

Table 6-5

Single Nuclide $DCGL_W$ Values for Detectable Radionuclides

Radionuclide	Peak of the Mean Dose (mrem/y per pCi/g)	$DCGL_W$ (pCi/g)
C-14	2.93E-06	8.33E+06
Co-60	1.93E+00	1.26E+01
Ni-63	1.60E-06	1.52E+07
Sr-90	3.76E-03	6.49E+03
Cs-134	1.09E+00	2.24E+01
Cs-137	4.62E-01	5.28E+01

6.6.2.6 Applicability of Surface Soil $DCGL_W$s to Sub-Surface Soil

Single nuclide $DCGL_W$ values for surface soil developed in Section 6.6.2.5 were developed for surface soil. However, subsurface soil (i.e., soil at depths greater than 15 centimeters (5.9 in) below the soil surface) contamination has been identified within the Industrial Area at Rancho Seco. Therefore, it is necessary to evaluate the applicability of the $DCGL_W$ values to subsurface soil contamination. This evaluation is performed in accordance to the guidance provided in Appendix I, Section 2.3.1 of NUREG-1757, Volume 2.

6.6.2.6.1 Radionuclide Concentration Values for Subsurface Soil Dose Calculations

The unity rule was applied to the mixture concentrations of detected radionuclides listed in Table 6-2 (decayed to July 1, 2008 to represent the radionuclide mixture at the approximate completion of final status surveys) using the single nuclide DCGL concentration values of Table 6-5 to calculate maximum radionuclide concentration limits that will result in an annual dose to the industrial worker under the industrial worker scenario of 25 millirem. The unity rule is defined in the following equation:

$$\frac{C_{M(C-14)}}{DCGL_{C-14}}+\frac{C_{M(Co-60)}}{DCGL_{Co-60}}+\frac{C_{M(Ni-63)}}{DCGL_{Ni-63}}+\frac{C_{M(Sr-90)}}{DCGL_{Sr-90}}+\frac{C_{M(Cs-134)}}{DCGL_{Cs-134}}+\frac{C_{M(Cs-137)}}{DCGL_{Cs-137}}\leq 1$$

Equation 6-1

where:

$C_{M(x)}$ = the mixture concentration of radionuclide "x"

Applying the decayed radionuclide concentrations from Table 6-2 and the single nuclide DCGL values from Table 6-5 and solving the unity rule equation results in the following:

$$\frac{4.74}{8.33E+06}+\frac{6.66}{1.26E+01}+\frac{170}{1.52E+07}+\frac{1.28}{6.49E+03}+\frac{0.230}{2.24E+01}+\frac{948}{5.28E+01}=18.5$$

Equation 6-2

Dividing both sides of Equation 2 by 18.5 to maintain the unity rule results in:

$$\frac{0.256}{8.33E+06}+\frac{0.360}{1.26E+01}+\frac{9.18}{1.52E+07}+\frac{0.0692}{6.49E+03}+\frac{0.0124}{2.24E+01}+\frac{51.2}{5.28E+01}=1$$

Equation 6-3

Results of the radionuclide mixture concentrations calculated in Equation 6-3 are provided in Table 6-6.

Table 6-6

Maximum Allowable Radionuclide Mixture Concentrations

Radionuclide	Mixture Conc. (pCi/g)	Radionuclide	Mixture Conc. (pCi/g)
C-14	2.56E-01	Sr-90	6.92E-02
Co-60	3.60E-01	Cs-134	1.24E-02
Ni-63	9.19E+00	Cs-137	5.12E+01

Use of these radionuclide concentrations for homogeneously mixed surface soil (top 15 centimeters) under the industrial worker scenario results in an annual dose to the industrial worker of 25 millirem.

6.6.2.6.2 Evaluation of Dose Effects from Varying Contamination Layer Thickness

Surface soil is described by the NRC in NUREG-1757, Volume 2, as the top layer of the soil, which is approximately 15 centimeters (5.9 in) thick. The single nuclide DCGLs calculated in Section 6.6.2.5 were based on a contamination layer thickness of 15 centimeters. Actual contaminated soil thickness, however, may be considerably deeper than the top 15 centimeters. Therefore, peak of the mean dose using the maximum allowable radionuclide concentrations of Table 6-6 was calculated by RESRAD v6.22 in the probabilistic mode with contaminated zone depths of 0.15, 0.5, 1, 1.5, 2, 2.5 and 3 meters. Deterministic input parameters are listed in Appendix 6-F and stochastic parameter statistical distributions are listed in Appendix 6-G. The initial contaminated zone parameters were retained for all depths while thickness of underlying uncontaminated zones was reduced to account for the increased contaminated zone thickness. Peak of the mean nuclide dose results are summarized in Table 6-7 and depicted graphically in Figure 6-6 and Figure 6-7.

Table 6-7

Peak of the Mean Dose vs Contaminated Layer Thickness

Nuclide	Dose (mrem/year) at Contaminated Layer Thickness						
	0.15 m	0.5 m	1.0 m	1.5 m	2.0 m	2.5 m	3.0 m
C-14	7.51E-07	2.35E-06	4.57E-06	6.83E-06	9.09E-06	1.13E-05	1.36E-05
Co-60	6.95E-01	8.12E-01	8.17E-01	8.17E-01	8.18E-01	8.18E-01	8.18E-01
Ni-63	1.48E-05	2.22E-05	2.23E-05	2.23E-05	2.23E-05	2.23E-05	2.23E-05
Sr-90	2.61E-04	2.91E-04	2.93E-04	2.94E-04	2.95E-04	2.95E-04	2.95E-04
Cs-134	1.35E-02	1.48E-02	1.49E-02	1.49E-02	1.49E-02	1.49E-02	1.49E-02
Cs-137	2.36E+01	2.57E+01	2.57E+01	2.57E+01	2.57E+01	2.57E+01	2.57E+01
Total	2.43E+01	2.65E+01	2.66E+01	2.66E+01	2.66E+01	2.66E+01	2.66E+01

As shown in the results, calculation of DCGLs based on surface soil (top 15 centimeters) is slightly non-conservative. There is a 9.05 percent increase in calculated total peak of the mean dose by increasing the contaminated layer thickness from 0.15 meters to 0.5 meters. However, there is little additional increase in total peak of the mean dose by increasing the contaminated layer thickness up to 3 meters. This non-conservatism may be discounted unless sub-surface soil contamination exists over a large area (greater than 300 m^2). At 300 m^2 the area factor for Cs-137 (the predominant dose contributor for the Table 6-6 radionuclide mixture) calculated in DTBD-05-003, "Soil and Structural Surface Area Factors for Use at Rancho Seco," [Reference 6-17] is 1.11, which is greater than the non-conservatism of 8.74 percent. The area factor for Cs-137 increases for areas less than 300 m^2 up to a factor of 11.3 for 1 m^2.

The steady dose increase with increasing contaminated layer thickness seen for C-14 is because the dominant exposure pathway for C-14 is inhalation. The C-14 available for inhalation is directly proportional to the total quantity of C-14 in the soil because C-14 is volatilized from sub-surface as well as surface soil.

6.6.2.6.3 Evaluation of Discrete Pockets of Contamination at Depth

Since subsurface soil contamination has only been observed to occur in discrete pockets, the application of surface soil DCGLs to subsurface pockets of contamination was evaluated. For purposes of evaluation, these discrete pockets were defined as cylindrical volumes of soil 100 m^2 on the surface and 2 m deep. The soil was considered to be contaminated to the maximum allowable concentrations listed in Table 6-6. Peak of the mean dose calculations were performed with the pocket exposed to the surface and 0.25, 0.5, 1, 2.5, 5 and 10 meters below the surface.

Deterministic input parameters are listed in Appendix 6-H and stochastic parameter statistical distributions are listed in Appendix 6-I. The calculations performed did not use the simplified mathematical model developed in Section 6.6.2.3. Because the discrete pockets of soil contamination transverse several different soil strata, generic stochastic statistical parameter distributions were used to represent the soil physical parameters rather than the deterministic sensitive soil parameters developed in Section 6.6.2.3. Peak of the mean dose results are summarized in Table 6-8 and depicted graphically in Figure 6-8. The times of the peak of the mean dose (years since performance of the FSS) are also included in Table 6-8 and depicted graphically in Figure 6-9.

Table 6-8

Peak of the Mean Dose vs Discrete Contamination Pocket Depth

	Dose (mrem/year) at Contamination Pocket Depth*						
	0 m	**0.25 m**	**0.5 m**	**1.0 m**	**2.5 m**	**5.0 m**	**10 m**
P-o-M♠	2.23E+01	1.54E+00	4.50E-01	2.79E-01	1.36E-01	6.00E-02	2.64E-02
P-o-M Time (y)	0.00E+00	5.38E+01	2.42E+01	2.42E+01	4.12E+01	5.38E+01	7.02E+01

*Depth of top of discrete contamination pocket below ground surface
♠Peak of the mean dose at time of peak of the mean dose

As shown in Table 6-8 and Figure 6-8, the peak of the mean dose decreases with increasing depth of the discrete pockets of contamination beneath the soil surface. Therefore, application of surface soil DCGL values to subsurface soil contamination is conservative. Although DCGL values for discrete pockets of subsurface soil contamination could be developed that are higher than the surface soil DCGL values, these subsurface soil DCGL values would be non-conservative if the subsurface soil contamination is excavated at some later date and spread on the surface, thus becoming surface soil contamination. Therefore, surface soil DCGL values should be applied to discrete pockets of subsurface soil contamination.

6.6.3 Structural Surfaces

6.6.3.1 Identification Of RESRAD-BUILD Sensitive Parameters

A theoretical site-specific suite of 26 radionuclides potentially present at Rancho Seco was derived in Section 6.3. Single nuclide DCGL values were derived for each radionuclide in this theoretical site-specific suite.

RESRAD-BUILD v3.22 was used to generate the single nuclide DCGL values based on an industrial worker building occupancy scenario introduced in NUREG/CR-6755, "Technical Basis for Calculating Radiation Doses for the Building Occupancy Scenario Using the Probabilistic RESRAD-BUILD 3.0 Code," [Reference 6-18]. The RESRAD-BUILD code is a pathway analysis model designed to evaluate the potential radiological dose to an individual who works in a building contaminated with radioactive material. It considers the releases of radionuclides into the indoor air by diffusion, mechanical removal, or erosion. RESRAD-BUILD considers seven exposure pathways:

1. External exposure directly from the source,
2. External exposure to materials deposited on the floor,
3. External exposure due to air submersion,
4. Inhalation of airborne radioactive particulates,
5. Inhalation of aerosol indoor radon progeny (in the case of the presence of radon predecessors) and tritiated water vapor (the radon pathway was turned off because the NRC does not regulate dose received from radon and progeny),
6. Inadvertent ingestion of radioactive material directly from the source, and
7. Ingestion of materials deposited on the surfaces of the building compartments.

A sensitivity analysis was performed first to identify the parameters that are sensitive in the industrial worker building occupancy scenario. As shown schematically in Figure 6-10, the parameter selection process starts with the selection of the RESRAD-BUILD parameter to be evaluated. The RESRAD-BUILD parameters were treated as described in Section 6.6.1.

The dose model included five contaminated surfaces, four walls and a floor. The ceiling was assumed to be either not contaminated, replaced if the room would be reused or to be so far above the floor that it would provide an insignificant contribution to dose to the receptor.

Deterministic input parameters are listed in Appendix 6-J and stochastic parameter statistical distributions are listed in Appendix 6-K. Once the parameter values listed in Appendix 6-J and the statistical parameter distributions listed in Appendix 6-K were loaded into RESRAD-BUILD v3.22 (v3.3 for Eu-154), the code was run in the probabilistic mode for each radionuclide of concern to identify the sensitive parameters. The absolute value of the calculated PRCC at time 1 was then used to classify the parameters with statistical distributions as sensitive or non-sensitive. PRCC was chosen because NUREG/CR-6692 recommends that it be used when nonlinear relationships, widely disparate scales or long tails are present in the inputs and outputs. If the absolute value of the PRCC was greater than 0.10, then the parameter was classified as sensitive. If the absolute value of the PRCC was equal to or less than 0.10, then the parameter was classified as non-sensitive. The sensitive parameters for each radionuclide and the sensitive parameter PRCC values are listed in Appendix 6-L.

6.6.3.2 Derivation Of Single Nuclide DCGL Values

Single nuclide DCGL values for the radionuclides of concern were calculated by performing an individual RESRAD-BUILD calculation for each radionuclide of concern. First, the applicable priority 2 and 3 parameter values listed in Appendix 6-M were loaded into RESRAD-BUILD v3.22 (v3.3 for Eu-154). Next, the non-sensitive priority 1 parameters were treated stochastically by loading the appropriate statistical parameter distributions listed in Appendix 6-K. Finally, the priority 1 sensitive parameters were treated deterministically by loading the assigned parameter value listed in Appendix 6-K and running the code in the probabilistic mode to identify the "Statistics for Dose" for Time 1 for each radionuclide of concern. For each calculation, the Latin Hypercube sampling technique was used with a random seed of 1000, 300 observations and one repetition.

Five parameter values that could be treated stochastically for certain radionuclides (not all radionuclides have the same sensitive parameters as shown in Appendix 6-L) were treated deterministically for all radionuclides. These parameters included room height, room floor area, deposition velocity and air exchange rate, which were assigned the sensitive parameter values listed in Appendix 6-K. Due to the large variation in room size in the structures expected to remain at Rancho Seco, the sensitive parameter values for room height and room floor area were selected to provide conservative calculations. The sensitive parameter values for deposition velocity and air exchange rate were selected to provide consistency in calculation assumptions.

The value of the fifth parameter, resuspension rate, was calculated based on the NUREG-1720, "Re-evaluation of the Indoor Resuspension Factor for the Screening Analysis of the Building Occupancy Scenario for NRC's License Termination Rule," [Reference 6-19] recommended DandD resuspension factor of 9.6E-07 m^{-1} and the sensitive parameter values for deposition velocity, air exchange rate and room height in accordance with the following equation (Equation 3.5 of NUREG/CR-6755):

$$\lambda_r = R_f \left(v_{dep} + \lambda_a \times H \right)$$

Equation 6-4

where

λ_r = the RESRAD-BUILD parameter, resuspension rate (DKSUS),

R_f = the DandD parameter, resuspension factor,

v_{dep} = the RESRAD-BUILD parameter, deposition velocity (UD),

λ_a = the RESRAD-BUILD parameter, air exchange rate (LAMBDATE), and

H = room height (H).

Radionuclide dose conversion factors (DCFs) were calculated by performing a probabilistic RESRAD-BUILD v3.22 calculation for each of the 26 radionuclides present in the theoretical site-specific suite. The DCFs selected (in units of mrem/yr per 100 dpm/100 cm^2) are the average total dose values for the receptor. Single nuclide DCGL values were then calculated by dividing the dose limit (25 mrem/yr) by the DCF value to give single nuclide DCGL values in units of dpm/100 cm^2. Results of these calculations and the associated DCF values are listed in Table 6-9. The structural surface DCGLs were developed in Rancho Seco DTBD-05-005, "DCGLs for RSNGS Activated and Volumetrically Contaminated Bulk Materials," [Reference 6-20].

Table 6-9

Calculated Structural Surface Single Nuclide DCFs and DCGLs

Radionuclide	Dose Conversion Factor (mrem/yr per dpm/100 cm^2)	DCGL (dpm/100 cm^2)
H-3	7.94E-08	3.15E+08
C-14	2.92E-06	8.56E+06
Na-22	1.47E-03	1.70E+04
Fe-55	7.31E-07	3.42E+07
Ni-59	3.13E-07	7.99E+07
Co-60	1.64E-03	1.52E+04
Ni-63	8.20E-07	3.05E+07
Sr-90	2.07E-04	1.21E+05
Nb-94	1.09E-03	2.29E+04
Tc-99	2.13E-06	1.17E+07
Ag-108m	1.13E-03	2.21E+04
Sb-125	3.13E-04	7.99E+04
Cs-134	1.14E-03	2.19E+04
Cs-137	4.50E-04	5.56E+04
Pm-147	1.50E-06	1.67E+07
Eu-152	7.86E-04	3.18E+04
Eu-154	8.43E-04	2.97E+04
Eu-155	4.78E-05	5.23E+05
Np-237	1.05E-02	2.38E+03
Pu-238	7.30E-03	3.42E+03
Pu-239	8.19E-03	3.05E+03
Pu-240	8.19E-03	3.05E+03
Pu-241	1.37E-04	1.82E+05
Am-241	8.37E-03	2.99E+03
Pu-242	7.81E-03	3.20E+03
Cm-244	4.15E-03	6.02E+03

The calculations for Eu-154 were performed using RESRAD-BUILD v3.3 after it was released by Argonne during the summer of 2005. RESRAD-BUILD v3.22 Eu-154 had a coefficient problem that related the FGR 13 dose conversion factor to source size and shield thickness incorrectly, resulted in an erroneous calculated dose conversion factor. This problem was corrected in RESRAD-BUILD v3.3.

6.6.4 Bulk Material

While the single nuclide DCGLs described in Section 6.6.3 are applicable to most structural surfaces, the potential exists that some structural surfaces are the face surfaces of structural components containing volumetric contamination arising from neutron activation. The possibility also exists for some volumetric contamination caused by the migration of surface contamination into the materials of construction. Therefore, it is necessary to calculate single nuclide DCGLs for bulk materials in order to evaluate these surfaces during the conduct of final status surveys. The bulk material DCGLs were developed in Rancho Seco DTBD-05-005, "DCGLs for RSNGS Activated and Volumetrically Contaminated Bulk Materials," [Reference 6-21].

6.6.4.1 Radionuclides of Concern for Activated Bulk Material Dose Calculations

A site-specific suite of potential radionuclides for use at Rancho Seco was derived in Section 6.3. This suite of potential radionuclides contains a total of 26 radionuclides. Single nuclide bulk material DCGL values were calculated for the entire suite of 26 radionuclides.

6.6.4.2 Identification of RESRAD-BUILD Sensitive Parameters for Bulk Material

Section 6.6.3 used RESRAD-BUILD v3.22 to generate single nuclide structural surface DCGL values based on an industrial worker building occupancy scenario introduced in NUREG/CR-6755. Section 6.6.3.1 identified sensitive parameters for RESRAD-BUILD v3.22 and established the dose model for derivation of DCGLs for structural surfaces. The dose model included five contaminated surfaces, four walls and a floor. The ceiling was assumed to be either not contaminated, replaced if the room would be reused or to be so far above the floor that it would provide an insignificant contribution of dose to the receptor. Due to the large variety of room sizes in the structures that will remain after license termination, the room dimensions were determined probabilistically and given conservative deterministic values.

Only portions of the Section 6.6.3 dose model are considered appropriate for derivation of single nuclide DCGL values for activated or volumetrically contaminated bulk material. Because most interior concrete in the containment building, down to the carbon steel liner plate, will be removed; only the carbon steel liner and concrete below it that are in the area formerly below the reactor vessel have a potential of being activated. Also, in other areas of the remaining structures the floors will have the highest possibility of containing volumetric contamination due to spills of radioactive liquids. Therefore, only the floor area of 137 m^2 derived in the Section 6.6.3 dose model will be used by replacing the floor surface source with a 1 foot thick (the most likely maximum depth of activation or contamination according to NUREG/CR-5884, Volume 2, Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station, [Reference 6-22]) volume source having the same face surface area as the Section 6.6.3 dose model floor source.

Other parameters that were found to be sensitive in Section 6.6.3.1 and treated deterministically for the derivation of structural surface single nuclide DCGLs may not remain sensitive for volume sources. Therefore, they were treated stochastically for the sensitivity analysis using volume sources.

For the case of tritium in the volume sources, the tritium was assumed to be present in the volume sources in the form of water that is released from the volume sources in the form of vapor (HTO vapor). Under this assumption, ANL/EAD/03-01, "User's Manual for RESRAD-BUILD Version 3," [Reference 6-23] recommends that the deposition velocity be treated deterministically and set to "0".

The same parameter selection process used in Section 6.6.3.1was used for this sensitivity analysis. The parameter values for sensitivity analysis and their assigned classification and priority are provided in Appendix 6-N. The parameter statistical distributions and sensitive parameter selection results for Rancho Seco priority 1 and 2 physical parameters are listed in Appendix 6-O. The parameter values for sensitivity analysis in Appendix 6-N and the parameter distributions listed in Appendix 6-O were loaded into RESRAD-BUILD v3.3 separately for each radionuclide.

Once the parameter values and the statistical parameter distributions were loaded into RESRAD-BUILD v3.3, the code was run in the probabilistic mode for each radionuclide of concern separately to identify the sensitive parameters for that radionuclide. For each calculation, the Latin Hypercube sampling technique was used with a random seed of 1000, 300 observations and one repetition. The absolute value of the calculated PRCC at time 1 was then used to classify the parameters with statistical distributions as sensitive or non-sensitive. If the absolute value of the PRCC was greater than 0.10, then the parameter was classified as sensitive. If the absolute value of the PRCC was equal to or less than 0.10, then the parameter was classified as non-sensitive.

Values for use in dose modeling for the physical parameters with sensitive parameters were selected based on sensitivity of the calculated PRCC following the guidance of NUREG/CR-6676. If the absolute value of the PRCC was greater than 0.10, then the parameter value at either the 75% quartile or the 25% quartile was selected based on TEDE correlation with the parameter. If the PRCC value was negative, the parameter to dose correlation is negative and the parameter value at the 25% quartile was selected. If the PRCC value was positive, the parameter to dose correlation is positive and the parameter value at the 75% quartile was selected. The sensitive parameter deterministic values and the highest sensitive parameter PRCC values (absolute value) of all radionuclides evaluated are listed in Appendix 6-O (with an accuracy of three significant figures).

6.6.4.3 Derivation Of Single Nuclide Bulk Material DCGL Values

Single nuclide dose conversion factors (DCFs) were calculated probabilistically, repeating the RESRAD-BUILD v3.3 calculations performed in Section 6.6.4.2 by replacing the stochastic parameter distributions for each radionuclide identified to be sensitive in Appendix 6-O with the assigned deterministic parameter value listed in Appendix 6-O. The DCF for each radionuclide was determined by performing these calculations with a source concentration of 1 pCi/gram to provide a DCF with the units of mrem/year per pCi/gram. The probabilistic dose used was the average total dose from the "Statistics for Dose (mrem) for Time: 1" report. The DCF for each radionuclide (with an accuracy of three significant figures) is provided in Table 6-10. DCGL values were then calculated by dividing the dose limit (25 mrem/yr) by the DCF value to give DCGL values in units of pCi/gram. Results of these calculations are also in provided in Table 6-10.

Table 6-10
Bulk Material Single Nuclide DCF and DCGL Values

Radionuclide	Dose Conversion Factor (mrem/yr per pCi/g)	DCGL (pCi/g)
H-3	3.18E-03	7.86E+03
C-14	1.56E-05	1.60E+06
Na-22	2.98E+00	8.39E+00
Fe-55	6.40E-07	3.91E+07
Ni-59	1.68E-06	1.49E+07
Co-60	3.54E+00	7.06E+00
Ni-63	3.65E-06	6.85E+06
Sr-90	6.01E-03	4.16E+03
Nb-94	2.11E+00	1.18E+01
Tc-99	3.39E-05	7.37E+05
Ag-108m	2.09E+00	1.20E+01
Sb-125	5.26E-01	4.75E+01
Cs-134	2.05E+00	1.22E+01
Cs-137	7.40E-01	3.38E+01
Pm-147	1.52E-05	1.64E+06
Eu-152	1.52E+00	1.64E+01
Eu-154	1.67E+00	1.50E+01
Eu-155	3.20E-02	7.81E+02
Np-237	3.34E-01	7.49E+01
Pu-238	6.92E-02	3.61E+02
Pu-239	2.04E-01	1.23E+02
Pu-240	8.45E-02	2.96E+02
Pu-241	1.22E-03	2.05E+04
Am-241	9.26E-02	2.70E+02
Pu-242	8.09E-02	3.09E+02
Cm-244	3.72E-02	6.72E+02

6.6.5 Containment Building Interior Surfaces

The District has no plans for beneficial reuse of the Rancho Seco Containment Building. Furthermore, remediation during the decommissioning process will have removed all equipment and structures from the containment building interior and the majority of interior concrete will be also be removed leaving only the carbon steel liner plate. Because of this, the industrial worker building occupancy scenario used to generate structural surface DCGLs in Section 6.6.3 is not a realistic scenario to be applied to the interior surface of the containment building after completion of remediation.

The final condition of the containment building will be one of no ventilation, lighting or power. Also, access will be extremely restricted with most containment penetrations, including personnel access hatches, closed off. Under these conditions, the occupancy factor will not provide the limiting condition for the derivation of containment building surface DCGLs with

RESRAD-BUILD using a normal mathematical model. An industrial worker building inspection scenario is developed in Section 6.6.5.1 and containment surface DCGLs were calculated for this scenario.

The District has no current plans to renovate or demolish the containment building, either prior to or after license termination; however, a building renovation/demolition scenario would be the most limiting scenario for the derivation of containment building surface DCGLs. Therefore, even though it is unlikely that the containment building will be renovated or demolished after license termination, a building renovation/demolition scenario can be used to derive Containment Building surface DCGLs for this scenario.

NUREG/CR-5512, Vol. 1, describes a building renovation scenario. It states:

> *"The building renovation scenario, in Section 3.1, accounts for an average volume (subsurface) concentration of radionuclides in building walls, floors and ceilings. ...*
>
> *...During renovation or demolition, surface and volume sources will be disturbed, creating loose contamination. This loose contamination can produce higher concentrations of radionuclides in the air or on surfaces than the levels in an undisturbed building.*
>
> *Renovation conditions serve as the prudently conservative basis for this scenario analysis. The differences between renovation and demolition are difficult to predict, but both can likely be represented by the same conceptual model. For some conditions, demolition may represent a worst-case situation; in others, renovation may be the worst case. ..."*

ANL/EAD/03-1 provides directions for the use of RESRAD-BUILD for various scenarios, including the building renovation scenario. The exposure pathways considered when using RESRAD-BUILD include:

1. External exposure to penetrating radiation emitted directly from the source,
2. External exposure to penetrating radiation emitted from radioactive particulates deposited on the floors of the compartments,
3. External exposure to penetrating radiation due to submersion in airborne radioactive particulates,
4. Inhalation of airborne radioactive particulates,
5. Inhalation of aerosol indoor tritiated water vapor,
6. Inadvertent ingestion of radioactive material contained in removable material directly from the source, and
7. Inadvertent ingestion of airborne radioactive particulates deposited on the surfaces of the building.

ANL/EAD/03-1 also provides an input data template and input parameter values for the building renovation scenario. The template and input parameter values are designed to match the building renovation scenario introduced in NUREG/CR-5512, Vol. 1. The calculations performed to derive containment building DCGLs used these parameter values. The exposure duration specified in ANL/EAD/03-1 is 179 days, which is the renovation period specified in NUREG/CR-5512, Vol. 1 for the building renovation scenario. Based on the Maine Yankee containment building demolition experience, this is a reasonable exposure duration.

6.6.5.1 Application of an Industrial Worker Building Inspection Scenario

6.6.5.1.1 Radionuclides of Concern for Industrial Worker Building Inspection Scenario DCGL Calculations

A list of nine significant radionuclides out of the 26 radionuclides in the site-specific suite of radionuclides based on characterization samples were identified in DTBD-05-015, Rancho Seco Nuclear Generating Station Structure Nuclide Fraction and DCGLs, [Reference 6-24]. These nine radionuclides include Co-60, Sr-90, Cs-134, Cs-137, Pu-238, Pu-239, Pu-240, Pu-241 and Am-241. Industrial worker building inspection scenario single nuclide DCGLs were derived for these nine radionuclides.

6.6.5.1.2 Mathematical Containment Building Model Used with RESRAD-BUILD

RESRAD-BUILD is designed to perform dose modeling on up to three rectangular compartments, each containing a floor, four walls and a ceiling. For purposes of modeling the Rancho Seco containment building, a single compartment was selected. The compartment is specified by defining a floor area and compartment height. The renovation/demolition scenario defined in ANL/EAD/03-1 specifies the use of volume sources in the mathematical model. Volume sources are modeled in RESRAD-BUILD as cylinders with the source direction defined as the vector from the face of the cylinder perpendicular to the exposed area. The industrial worker building inspection scenario utilizes surface sources with the same areas as the volume sources used in the renovation/demolition scenario.

The Rancho Seco containment building is a cylindrical structure containing a circular basemat, cylindrical walls and a domed ceiling. Therefore, several simplifying assumptions were necessary to model the containment building to obtain a mathematical model usable by RESRAD-BUILD. For use by RESRAD-BUILD, the floor and ceiling were modeled as circular sources 130-feet in diameter to match the containment building interior diameter. The floor source represented the containment basement floor and the ceiling source represented the base of the dome located 145-feet above the center of the basement floor. Next, the cylindrical vertical wall area (59,400 ft^2) was divided into four equal areas and each fourth of the area assigned as the area of each of the four wall sources. This is considered an acceptable approximation of the containment building interior because the majority of calculated potential dose to the renovation/demolition worker is from inhalation and ingestion of the dust created during demolition. This is proportional to total volume, not the shape of the containment building interior surfaces.

6.6.5.1.3 Derivation Of Industrial Worker Building Inspection Scenario Single Nuclide DCGL Values

Industrial worker building inspection scenario single nuclide DCGLs were derived deterministically using the parameters provided in Appendix 6-P. The input parameters contained in Appendix 6-P were input into the RESRAD-BUILD v3.3 code for each of the nine radionuclides of concern. The DCFs selected (in units of mrem/yr per 100 dpm/100 cm^2) are the average total dose values for the receptor. Single Nuclide DCGL values were then calculated by dividing the dose limit (25 mrem/yr) by the DCF value to give DCGL values in units of dpm/100 cm^2. Results of these calculations are provided in Table 6-11 below.

Table 6-11

Containment Building Surface Single Nuclide DCF and DCGL Values for the Industrial Worker Building Inspection Scenario

Radionuclide	Dose Conversion Factor (mrem/yr per 100 dpm/100 cm^2)	DCGL (dpm/100 cm^2)
Co-60	2.81E-05	8.90E+05
Sr-90	1.46E-05	1.71E+06
Cs-134	2.38E-05	1.05E+06
Cs-137	1.09E-05	2.29E+06
Pu-238	3.10E-04	8.06E+04
Pu-239	3.43E-04	7.29E+04
Pu-240	3.43E-04	7.29E+04
Pu-241	6.63E-06	3.77E+06
Am-241	3.53E-04	7.08E+04

6.6.5.2 Application of a Building Renovation/Demolition Scenario

6.6.5.2.1 Radionuclides of Concern for Containment Building DCGL Calculations

The derivation of a theoretical site-specific suite of 26 radionuclides potentially present at RSNGS was discussed in Section 6.3. Single nuclide DCGL values were derived for each radionuclide in this theoretical site-specific suite.

6.6.5.2.2 Identification of RESRAD-BUILD Sensitive Parameters for Containment Building Renovation/Demolition

Sensitive parameters for containment building renovation/demolition were identified using the process described in Section 6.6.4.2. The parameter values for sensitivity analysis and their assigned classification and priority are provided in Appendix 6-Q. The parameter statistical distributions and sensitive parameter selection results for Rancho Seco priority 1 and 2 physical parameters are listed in Appendix 6-R.

Once the parameter values and the statistical parameter distributions were loaded into RESRAD-BUILD v3.3, the code was run in the probabilistic mode for each radionuclide of concern to identify the sensitive parameters for that radionuclide. For each calculation, the Latin Hypercube sampling technique was used with a random seed of 1000, 300 observations and one repetition. The absolute value of the calculated PRCC at time 1 was then used to classify the parameters with statistical distributions as sensitive or non-sensitive.

Values for use in dose modeling for the physical parameters with sensitive parameters were selected based on sensitivity of the calculated PRCC following the guidance of NUREG/CR-6676. If the absolute value of the PRCC was greater than 0.10, then the parameter value at either the 75% quartile or the 25% quartile was selected based on TEDE correlation with the parameter. If the PRCC value was negative, the parameter to dose correlation is negative and the parameter value at the 25% quartile was selected. If the PRCC value was positive, the parameter to dose correlation is positive and the parameter value at the 75% quartile was selected. The sensitive parameter deterministic values and the highest sensitive parameter PRCC values (absolute value) of all radionuclides evaluated are listed in Appendix 6-R (with an accuracy of three significant figures).

6.6.5.3 Derivation Of Containment Building Renovation/Demolition Single Nuclide DCGL Values

Single nuclide DCFs were calculated deterministically if all parameters treated stochastically in Section 6.6.5.2.2 were determined to be sensitive. Single nuclide DCFs were calculated probabilistically if some non-sensitive parameters could be treated stochastically. The RESRAD-BUILD v3.3 calculations performed in Section 6.6.5.2.2 were repeated by replacing the stochastic parameter distributions for each radionuclide identified to be sensitive in Appendix 6-R with the assigned deterministic parameter value listed in Appendix 6-R.

The DCF for each radionuclide was determined by performing the above calculations with a source concentration of 1,000 dpm/gram to provide a DCF with the units of mrem/year per 1,000 dpm/gram. The probabilistic dose used was the average total dose from the "Statistics for Dose (mrem) for Time: 1" report. The DCF for each radionuclide (with an accuracy of three significant figures) is provided in Table 6-12.

The building renovation scenario, also applicable to building demolition (see Section 6.6.5), described in NUREG/CR-5512, Vol. 1, and the input data template and input parameter values provided in ANL/EAD/03-1, specify the use of a volume source with a thickness of 15 cm. In the case of the containment building any residual contamination will likely be fixed on the interior surface rather than dispersed throughout the 15 cm thickness. If the assumption is made that containment building surface activity would be mixed into the 15 cm thickness during demolition, then DCGL values may be calculated by assuming that all of the activity contained in the source is actually on the surface and solving Equation 6-5 to give DCGL values in units of dpm/100 cm^2.

$$\frac{10^3}{DCF_i}\frac{dpm/g}{mrem}\times 25\ mrem\times 2.5\ g/cm^3\times 1500\ cm^3\ \ per\ one\ hundred\ \ cm^2 \quad = \quad DCGL_i\ dpm/100\ cm^2$$

Equation 6-5

Results of these calculations are also in provided in Table 6-12.

Table 6-12

**Containment Building Surface Single Nuclide
DCF and DCGL Values – Renovation/Demolition Scenario**

Radionuclide	Dose Conversion Factor (mrem/yr per 10^3 dpm/g)	DCGL (dpm/100 cm^2)
H-3	7.72E-02	1.21E+09
C-14	4.62E-01	2.03E+08
Na-22	1.98E+03	4.73E+04
Fe-55	1.50E-01	6.25E+08
Ni-59	6.64E-02	1.41E+09
Co-60	2.33E+03	4.02E+04
Ni-63	1.73E-01	5.42E+08
Sr-90	4.67E+01	2.01E+06
Nb-94	1.42E+03	6.60E+04
Tc-99	3.93E-01	2.39E+08
Ag-108m	1.44E+03	6.51E+04
Sb-125	3.56E+02	2.63E+05
Cs-134	1.40E+03	6.70E+04
Cs-137	5.15E+02	1.82E+05
Pm-147	5.46E-01	1.72E+08
Eu-152	1.02E+03	9.19E+04
Eu-154	1.11E+03	8.45E+04
Eu-155	2.14E+01	4.38E+06
Np-237	5.48E+03	1.71E+04
Pu-238	3.86E+03	2.43E+04
Pu-239	4.23E+03	2.22E+04
Pu-240	4.23E+03	2.22E+04
Pu-241	8.14E+01	1.15E+06
Am-241	4.38E+03	2.14E+04
Pu-242	4.05E+03	2.31E+04
Cm-244	2.44E+03	3.84E+04

6.6.5.4 Application of Containment Building DCGLs

It is clear by comparing the single nuclide DCGL values contained in Tables 6-11 and 6-12 that the containment building renovation/demolition scenario is the most conservative and that the single nuclide DCGL values contained in Table 6-12 should be applied to the containment building interior surfaces. However, a conservative approach will be imposed and the structural surface DCGLs derived in Section 6.6.3 will be applied to the reasonably accessible surfaces of the containment building. These surfaces will include the floor and wall surfaces up to the level of the polar crane rail. The containment building renovation/demolition single nuclide DCGLs will be applied above the polar crane rail and containment building dome surfaces in consideration of worker safety during remediation and FSS activities.

6.6.6 Buried Piping

Approximately 30,700 linear feet of buried pipe have been identified that is expected to remain at Rancho Seco after license termination. The buried pipe ranges from one inch I.D. to 108 inch I.D. and is associated with systems such as the nitrogen gas system (one inch I.D.) to the main circulating water system (108 inch I.D.). Buried piping that will remain following license termination is located at a soil depth of three or more feet. A majority of the buried piping that is associated with systems that contained known contamination has been excavated during decommissioning and piping systems remaining have a low potential for significant internal contamination.

Evaluation of the buried piping scenario utilized soil DCGL values derived in Section 6.6.2. Under the scenario, buried piping, contaminated on the interior surface, is assumed to disintegrate instantaneously upon license termination. The disintegrated media is assumed to be subsurface soil and the media volume is assumed to be equal to the piping volume with the contamination uniformly mixed in the soil volume. A gross DCGL value to apply to interior piping surface was derived using standard computational methods assuming the disintegrated media is contaminated to soil DCGL concentrations using average observed nuclide fractions for soil and piping surface contamination.

The calculations assumed an average radionuclide mixture of 0.17 for Co-60 and 0.83 for Cs-137 (95% C. L.). A conservative gross DCGL of 100,000 dpm/100 cm^2 on the interior surface of the buried pipe was found acceptable based upon these calculations. The details of this analysis were developed in Rancho Seco DTBD-05-013, "Buried Piping Scenario and DCGL Determination Basis," [Reference 6-25].

6.6.7 Embedded Piping

Approximately 5,360 linear feet of embedded pipe have been identified that will remain at Rancho Seco. The embedded pipe ranges from 0.75 inch I.D. to 18 inch I.D. and is associated with the Turbine Building, Auxiliary Building, Reactor Building, and Fuel Building drains. Embedded pipe is located at the drain entrance down to depths between 9 to 30 inches or more beneath the concrete surface, depending on the building.

The embedded piping scenario assumes that the piping remains in place following decommissioning and that the dose to the industrial worker is from direct gamma exposure from the residual activity in the pipe with allowance made for photon attenuation by the wall or floor thickness of concrete remaining over the pipe. Whole body dose from the embedded pipe will be considered additive along with the dose to the industrial worker resulting from residual activity on the walls or floors of the room or area in which the embedded pipe is present. The surface DCGL will be reduced by the dose contribution from the embedded piping in order to ensure compliance with the annual dose limit.

Embedded pipe is partially shielded and constrained by the encasing concrete that limits the dose to the industrial worker to that arising from the gamma emitters in the nuclide mixture. The impact of nuclides that are not gamma emitters is minimal because the pipe is not easily extracted nor is the interior surface readily accessible through the overlying concrete. A total of 53 samples were collected and analyzed by gamma spectroscopy from various drains, sumps, and trenches in the buildings previously mentioned. Twenty samples were selected that reflect the different piping systems covered by the 53 samples. In many instances, several samples were collected from one system. The radionuclide analyses indicated that the primary contributors to whole body dose are Cs-137 and Co-60. The Fuel Building pipe sample results indicate the presence of a small portion of non-gamma emitters in the nuclide fraction. The mean nuclide fractions for Cs-137 and Co-60, as determined by the 20 samples, were 0.802 and 0.161 respectively. The individual building mean fractions were within two standard deviations of the overall mean values indicating a consistent nuclide ratio. This compares well with the concrete nuclide fractions of 0.89 and 0.11 for Cs-137 and Co-60 respectively.

A conservative gross DCGL of 100,000 dpm/100 cm^2 on the interior surface of the embedded pipe was evaluated and found acceptable. MicroShield® runs were used to model the gamma exposure at one meter from the concrete surface resulting from 100,000 dpm/100 cm^2 (4.5E-4 μCi/cm^2) in the maximum size pipe in a given building one meter from the surface of the concrete covering the embedded pipe. The amount of the concrete shielding included in the model was based on the thinnest concrete covering above the largest diameter embedded pipe for the given building as determined from site drawings. An occupancy factor of 2,000 hours per year was assumed to calculate the annual dose rate. Results are shown in Table 6-13 below. The annual dose rates are all less than 1 percent of the 25 mrem/y annual limit. The details of this analysis were developed in Rancho Seco DTBD-05-009, "Embedded Piping Scenario and DCGL Determination Basis," [Reference 6-26].

Table 6-13
Embedded Pipe Annual Dose Rate By Building

Building	Turbine	Fuel	Auxiliary	Reactor
Max Pipe Size (inches)	8	8	6	18
Concrete Depth (inches)	18	30	9	12
Annual Dose Rate (mrem/y)	0.01	0.0002	0.19	0.12

The potential for the removal of the embedded pipe and consequent dose to an industrial worker at some time in the future was examined even though this was not part of the industrial worker building occupancy scenario. The published source of dose factors that came the closest to a pipe cutting and removal scenario was NUREG-1640, Volume 1, "Radiological Assessments for Clearance of Materials from Nuclear Facilities," [Reference 6-27]. If the mean dose factors (NUREG-1640, Volume 1, Table 3.24) and scenario for converting pipe into scrap material as outlined in NUREG-1640 are employed using a DCGL of 100,000 dpm/100 cm^2 and the given nuclide fraction for embedded pipe, the annual dose rates are calculated to be 4.0 mrem/y for Cs-137 and 2.7 mrem/y for Co-60. The dose contribution from Cs-137 was principally from the release of contamination and subsequent inhalation by the worker while the dose from Co-60 was mostly the whole body dose from handling the contaminated pipe. In order to preclude the additional dose contribution from embedded pipe, RSNGS plans to grout piping which has residual contamination above the adjusted NRC screening levels (Table 5.19 of NUREG-5512, Volume 3) of 20,000 dpm/100 cm^2. This action level limits the dose rate to the reclamation worker to 0.55 mrem/y from Co-60 and 0.79 mrem/y from Cs-137.

6.7 Derivation of Area Factors

As stated in NUREG-1757, Volume 2, the $DCGL_W$ is the average concentration across an area that is calculated to result in the average member of the critical group receiving a dose at the appropriate dose limit. The general assumption is that the concentration of the radionuclides in the source is fairly homogenous. The degree to which any single localized area can be elevated above the average, assuming the average is at the $DCGL_W$, and not invalidate the homogenous assumption is characterized by the $DCGL_{EMC}$. One method for determining values for the $DCGL_{EMC}$ is to modify the $DCGL_W$ using a correction factor that accounts for the difference in area and the resulting change in dose. The area factor is then the magnitude by which the concentration within the small area of elevated activity can exceed $DCGL_W$ while maintaining compliance with the release criterion.

An area factor for use in elevated measurement comparison during final status surveys is defined by Equation 6-6.

$$Area\ Factor = \frac{DCGL_{EMC}}{DCGL_W}$$

Equation 6-6

where:

$DCGL_W$ = Baseline average DCGL value, and
$DCGL_{EMC}$ = Elevated measurement comparison DCGL value

NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys," [Reference 6-28] provides the methodology for calculating area factors in Chapter 8. Chapter 8 states that the area factors should be calculated using dose pathway models and assumptions that are consistent with those used to calculate the $DCGL_W$. Area factors are computed by taking the ratio of the dose per unit concentration calculated by RESRAD or RESRAD-BUILD for the baseline area to that calculated for various smaller areas.

6.7.1 Area Factors for Rancho Seco Surface Soils

6.7.1.1 Radionuclides of Concern for Surface Soils

A site-specific suite of potential radionuclides for use at Rancho Seco was derived in Section 6.3. Of the suite of 26 potential radionuclides, only six radionuclides were positively identified. These were C-14, Co-60, Ni-63, Sr-90, Cs-134, and Cs-137. Single nuclide DCGL concentration values (each radionuclide DCGL concentration represents 25 millirem per year) were derived for a baseline default area of 10,000 m^2 in Section 6.6.2 for each of the six detected radionuclides. These single nuclide DCGL concentration values are provided in Table 6-14 below. Area factors are calculated in this section only for the six radionuclides for which Section 6.6.2 derived DCGLs.

Table 6-14

Single Nuclide DCGL Values for Detectable Radionuclides

Radionuclide	Peak of the Mean Dose (mrem/y per pCi/g)	DCGL (pCi/g)
C-14	2.93E-06	8.33E+06
Co-60	1.93E+00	1.26E+01
Ni-63	1.60E-06	1.52E+07
Sr-90	3.76E-03	6.49E+03
Cs-134	1.09E+00	2.24E+01
Cs-137	4.62E-01	5.28E+01

6.7.1.2 Mathematical Hydrogeological Model

The mathematical hydrogeological model developed in Section 6.6.2.1 was used to calculate area factors for surface soils.

6.7.1.3 Calculation of Dose to Source Ratios for Surface Soil Area Factors

Dose to source ratios (DSRs) for the detectable radionuclides of concern were calculated by performing individual RESRAD probabilistic calculations for each of the six detectable radionuclides for each of nine specified contaminated area sizes. The site-specific RESRAD v6.22 dose model was first configured with the simplified mathematical model parameters contained in Appendix 6-S then with the statistical parameter distributions provided in Appendix 6-T. Sensitive parameters identified in Section 6.6.2.3 (density of the contaminated zone, contaminated zone K_d value for Cs-137 and external gamma shielding factor) were treated deterministically using the sensitive parameter values listed in Appendix 6-S. Parameters that were not sensitive were treated stochastically using the statistical parameter distributions contained in Appendix 6-T. RESRAD was then run in the probabilistic mode for each detected radionuclide and for each of the nine specified contaminated area sizes. A new value for the parameter “length of contaminated zone parallel to the aquifer flow” was used each time the contaminated area size was changed. The uncertainty analysis input settings for these calculations were:

- Latin Hypercube sampling,
- Random seed – 1000,
- Number of observations – 300,
- Number of repetitions – 1, and
- Grouping of observations – correlated or uncorrelated.

These calculations provided the peak of the mean DSR in mrem/year per pCi/g for each detected radionuclide. These DSRs are listed in Table 6-15.

Table 6-15

Calculated Peak-of-the-Mean DSR Values

Contaminated Area (m^2)	Radionuclide DSR (millirem/year per pCi/gram)					
	C-14	**Co-60**	**Ni-63**	**Sr-90**	**Cs-134**	**Cs-137**
10,000	2.92E-06	1.93E+00	1.60E-06	3.76E-03	1.09E+00	4.62E-01
3,000	1.76E-06	1.89E+00	1.60E-06	3.69E-03	1.07E+00	4.53E-01
1,000	1.16E-06	1.85E+00	1.60E-06	3.62E-03	1.05E+00	4.44E-01
300	6.08E-07	1.72E+00	4.90E-07	3.13E-03	9.77E-01	4.15E-01
100	3.63E-07	1.56E+00	1.72E-07	2.77E-03	8.86E-01	3.76E-01
30	2.15E-07	1.19E+00	5.94E-08	2.12E-03	6.88E-01	2.92E-01
10	1.34E-07	8.09E-01	2.64E-08	1.44E-03	4.71E-01	2.00E-01
3	7.04E-08	3.82E-01	1.40E-08	6.82E-04	2.23E-01	9.48E-02
1	3.84E-08	1.63E-01	9.78E-09	2.94E-04	9.65E-02	4.09E-02

6.7.1.4 Calculation of Surface Soil Area Factors

The DSRs calculated in Section 6.7.1.3 were then used to calculate area factors in accordance with Equation 6-7.

$$AF_i = \frac{DSR_{10,000\,m^2}}{DSR_i}$$

Equation 6-7

where:

AF_i = Area Factor at EMC area i

$DSR_{10,000\,m^2}$ = DSR at the baseline area of 10,000 m^2

DSR_i = DSR for EMC area i

The results of these calculations are listed in Table 6-16, shown graphically in Figure 6-11 for gamma emitters and shown graphically in Figure 6-12 for beta emitters.

Table 6-16

Calculated Surface Soil Area Factors

Contaminated Area (m^2)	Radionuclide Area Factor (unitless)					
	C-14	Co-60	Ni-63	Sr-90	Cs-134	Cs-137
10,000	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
3,000	1.66E+00	1.02E+00	1.00E+00	1.02E+00	1.02E+00	1.02E+00
1,000	2.52E+00	1.04E+00	1.00E+00	1.04E+00	1.04E+00	1.04E+00
300	4.80E+00	1.12E+00	3.27E+00	1.20E+00	1.12E+00	1.11E+00
100	8.04E+00	1.24E+00	9.30E+00	1.36E+00	1.23E+00	1.23E+00
30	1.36E+01	1.62E+00	2.69E+01	1.77E+00	1.58E+00	1.58E+00
10	2.18E+01	2.39E+00	6.06E+01	2.61E+00	2.31E+00	2.31E+00
3	4.15E+01	5.05E+00	1.14E+02	5.51E+00	4.89E+00	4.87E+00
1	7.60E+01	1.18E+01	1.64E+02	1.28E+01	1.13E+01	1.13E+01

6.7.2 Area Factors for Rancho Seco Structural Surfaces

6.7.2.1 Radionuclides of Concern for Structural Surfaces

Section 6.3 identified a site-specific suite of 26 radionuclides as potentially present at Rancho Seco. Offsite laboratory analysis of characterization samples representing structural surfaces taken to date have not identified five of these radionuclides as being present above analytical minimum detectable activity (MDA) levels. In addition, three more radionuclides were identified to have activity levels above MDA in only one characterization sample and with questionable results, indicating that these might be false positive results. Therefore, a total of eight radionuclides were discounted from further area factor evaluation. These eight radionuclides include Na-22, Sb-125, Eu-152, Eu-154, Eu-155, Np-237, Pu-242 and Cm-244. Additional radionuclides were removed from consideration based on their dose factor or nuclide fraction. The remaining radionuclides considered for structural surface area factors include:

Co-60 Sr-90 Cs-134 Cs-137

6.7.2.2 Calculation of Dose to Source Ratios for Structural Surface Area Factors

Single nuclide DCGL values for Rancho Seco structural surfaces were developed in Section 6.6.3 using the probabilistic features of RESRAD-BUILD v3.22. These calculations were based on an industrial worker building occupancy scenario introduced in NUREG/CR-6755. Section 6.6.3.1 identified sensitive parameters for RESRAD-BUILD v3.22 and established the dose model for derivation of DCGLs for structural surfaces. The dose model included five contaminated surfaces; four walls and a floor. The room dimensions were determined probabilistically and given conservative deterministic values. Since area factors apply only to one contiguous surface, the floor was selected from which to derive area factors because it was the largest single surface area (137 m^2). The room dimensions and remaining deterministic and stochastic parameters were left as they were derived in Section 6.6.3.1. The RESRAD-BUILD v3.22 parameters used to develop area factors are provided in Appendix 6-U and the statistical parameter distributions for the parameters treated stochastically are provided in Appendix 6-V.

Area factors were calculated in increments ranging from the 137 m^2 floor area down to 0.5 m^2. Complete sets of area factors were calculated only for the principle gamma emitting radionuclides of Co-60, Cs-134 and Cs-137. Area factors for Sr-90 was calculated only for the 0.5 m^2 area to demonstrate that the area factors are conservatively bounded by the area factors calculated for the principle gamma emitting radionuclides.

Dose to source ratios in units of millirem/year per dpm/100 cm^2 were calculated first using RESRAD-BUILD v3.22 in the probabilistic mode to identify the "Statistics for Dose" for Time 1 to obtain the calculated DSR. For each calculation, the Latin Hypercube sampling technique was used with a random seed of 1000, 300 observations and one repetition. The DCFs selected (in units of mrem/yr per 100 dpm/100 cm^2) are the average total dose values for the receptor and are provided in Table 6-17.

Table 6-17

Calculated Mean DSR Values for Structural Surface Area Factors

Contaminated Area (m^2)	Radionuclide DSR (millirem/year per dpm/100 cm^2)			
	Co-60	Sr-90	Cs-134	Cs-137
137	1.19E-03	8.92E-05	8.07E-04	3.09E-04
68	9.75E-04	—	6.51E-04	2.45E-04
36	7.87E-04	—	5.21E-04	1.94E-04
25	6.83E-04	—	4.52E-04	1.67E-04
16	5.63E-04	—	3.71E-04	1.37E-04
9	4.21E-04	—	2.77E-04	1.02E-04
4	2.56E-04	—	1.68E-04	6.16E-05
1	8.60E-05	—	5.64E-05	2.07E-05
0.5	4.60E-05	4.26E-07	3.01E-05	1.10E-05

6.7.2.3 Calculation of Structural Surface Area Factors

The DSRs calculated in Section 6.7.2.2 were then used to calculate structural surface area factors in accordance with Equation 6-8.

$$AF_i = \frac{DSR_{137\,m^2}}{DSR_i}$$

Equation 6-8

where:

AF_i = Area Factor at EMC area *i*

$DSR_{137\,m^2}$ = DSR at the floor area of 137 m^2

DSR_i = DSR for EMC area *i*

The results of these calculations are listed in Table 6-18 and shown graphically in Figure 6-13.

Table 6-18

Calculated Surface Area Factor Values

Contaminated Area (m^2)	Radionuclide Area Factor			
	Co-60	Sr-90	Cs-134	Cs-137
137	1.00E+00	1.00E+00	1.00E+00	1.00E+00
68	1.22E+00	—	1.24E+00	1.26E+00
36	1.51E+00	—	1.55E+00	1.59E+00
25	1.74E+00	—	1.79E+00	1.85E+00
16	2.11E+00	—	2.18E+00	2.26E+00
9	2.83E+00	—	2.91E+00	3.03E+00
4	4.65E+00	—	4.80E+00	5.02E+00
1	1.38E+01	—	1.43E+01	1.49E+01
0.5	2.59E+01	2.09E+02	2.68E+01	2.81E+01

6.8 Comparison of Alternative Exposure Scenarios for Impacted Area Soils

Single nuclide DCGL values for impacted area soils were calculated in Section 6.6.2.5 for radionuclides detected in surface soils using an industrial worker scenario. Although the industrial worker scenario is considered the most likely scenario for the Rancho Seco site and is the scenario for this License Termination Plan submittal, the dose impact from using a resident farmer scenario at specified times following license termination (partial site release) and the dose impact of maintaining an industrial area but allowing cattle grazing within impacted areas outside the industrial area and consumption of meat from the grazing cattle by an offsite member of the public has been evaluated.

6.8.1 Radionuclides of Concern and Concentrations for Alternative Scenario Dose Calculations

A site-specific suite of potential radionuclides for use at Rancho Seco was derived in Section 6.3. This suite of potential radionuclides contained a total of 26 radionuclides. On May 28, 2004 Rancho Seco submitted a spent fuel pool cooler pad soil sample collected on March 8, 2004 to General Engineering Laboratories (GEL) for analysis of the entire suite of potentially present 26 radionuclides. This sample was known by onsite gamma spectroscopy to have the highest level of contamination of any soil samples collected during the site characterization process. Of the suite of 26 potential radionuclides, GEL positively identified only six radionuclides. These were C-14, Co-60, Ni-63, Sr-90, Cs-134, and Cs-137. Single nuclide DCGL concentration values were Section 6.6.2.5 for each of the six radionuclides detected by GEL. These single nuclide DCGL concentration values are provided in Table 6-5. The GEL measured activity concentrations; decayed from the date of analysis to July 1, 2008 to represent the radionuclide mixture at the approximate completion of final status surveys were provided in Table 6-2. The unity rule was applied to a mixture of the detected radionuclides in Section 6.6.2.6.1 using the single nuclide DCGL concentration values of Table 6-5 to derive a radionuclide mixture that will result in an annual dose to the industrial worker under the industrial worker scenario of 25 millirem. The concentration values for this mixture were provided in Table 6-6.

6.8.2 Resident Farmer Alternative Exposure Scenario

6.8.2.1 Sensitivity Analysis of Detectable Radionuclides for the Resident Farmer Scenario

A sensitivity analysis was performed first to identify the parameters that are sensitive in the resident farmer scenario for the detectable radionuclides following the same methodology as that used in Section 6.6.2.3. The parameters selected for sensitivity analysis and their selection justification are provided in Appendix 6-W and the statistical parameter distributions used are provided in Appendix 6-X.

Once the parameter values listed in Appendix 6-W and the statistical parameter distributions listed in Appendix 6-X were loaded into RESRAD v6.22, the code was run in the uncorrelated parameter probabilistic mode for the radionuclides and concentrations provided in Table 6-6 to identify the sensitive parameters. This approach identified the sensitive parameters for the entire mixture using the resident farmer scenario, not just individual radionuclide parameter sensitivity. The uncertainty analysis input settings for this calculation were:

- Latin Hypercube sampling,
- Random seed – 1000,
- Number of observations – 300,
- Number of repetitions – 1, and
- Grouping of observations – correlated or uncorrelated.

The absolute value of the calculated partial ranked correlation coefficient (PRCC) of the peak of the mean dose was then used to classify the parameters with statistical distributions as sensitive or non-sensitive. The calculated PRCC values for both sensitive and non-sensitive parameters are listed in Appendix 6-X. If the absolute value of the PRCC was greater than 0.25, then the parameter was classified as sensitive. If the absolute value of the PRCC was equal to or less than 0.25, then the parameter was classified as non-sensitive. The sensitive parameters for each radionuclide and the sensitive parameter PRCC values are listed in Appendix 6-X.

Finally, values for use in dose modeling for the physical parameters with sensitive parameters were selected based on sensitivity of the calculated PRCC following the guidance of NUREG/CR-6676. If the absolute value of the PRCC was greater than 0.25, then the parameter value at either the 75% quartile or the 25% quartile was selected based on TEDE correlation with the parameter. If the PRCC value of the peak of the mean dose was negative, the parameter to dose correlation is negative and the parameter value at the 25% quartile was selected. If the PRCC value was positive, the parameter to dose correlation is positive and the parameter value at the 75% quartile was selected.

The sensitive parameters identified under the resident farmer scenario for detected radionuclides, in decreasing order of importance, include:

1. External gamma shielding factor,
2. Plant transfer factor for Cs,
3. Depth of roots,

4. Density of contaminated zone, and
5. Meat transfer factor for Cs.

6.8.2.2 Calculated Dose from Detected Radionuclides for a Resident Farmer Scenario

Once the sensitive parameters were identified for the resident farmer scenario, the parameter sensitivity model was revised by replacing the sensitive parameter statistical distributions with the deterministically assigned sensitive parameter values and running the model to calculate dose under the resident farmer scenario for industrial worker scenario maximum allowable radionuclide soil concentrations from Table 6-6. The dose factor library used for this calculation was the "RSNGS RF DCGL Dose" library containing the sensitive parameter values for plant, meat and milk transfer factors. The dose calculation times specified were 0, 1, 3, 25, 50, 75, 100, 300, 500 and 1000 years. This calculation assumes uniform contamination over the entire impacted area of the site and assumes that there has been no ALARA reduction from soil contamination limits. The calculated mean dose for the detected radionuclides at the specified times since license termination is provided in Table 6-19.

6.8.2.3 Calculated Dose from Discounted Radionuclides for a Resident Farmer Scenario

6.8.2.3.1 Discounted Radionuclides

As discussed in Section 6.6.2.2, only six of the potential radionuclides derived in Section 6.3 are considered to be radionuclides of concern for impacted area surface soil. This leaves 20 discounted radionuclides to be evaluated for potential dose using the resident farmer scenario. The potential dose is based on the minimum detectable activity (MDA) value for each radionuclide decayed to a FSS date of July 1, 2008 as discussed in Section 6.6.2.4. The decay corrected MDA concentrations were provided in Table 6-4.

6.8.2.3.2 Identification of Discounted Radionuclide Sensitive Parameters

In order to calculate potential dose from discounted radionuclides, the discounted radionuclides were first evaluated to identify sensitive RESRAD parameters to be treated deterministically in the dose calculation. The site-specific mathematical model developed in Section 6.6.2.1 was used with uncorrelated parameter distributions to perform the sensitivity analysis. The RESRAD v6.22 parameters used for the analysis are provided in Appendix 6-Y and the probabilistic parameter distributions and sensitivity analysis results are provided in Appendix 6-Z. The sensitivity analyses PRCC value results and the deterministic parameter selected for sensitive parameters are also included in Appendix 6-Z.

The sensitive parameters identified under the resident farmer scenario for discounted radionuclides in decreasing order of importance, include:

1. Depth of roots,
2. Plant transfer factor for Np,
3. Plant transfer factor for Tc,
4. Plant transfer factor for Pu,
5. External gamma shielding factor,
6. K_d of Tc-99 in contaminated zone

7. Fruit, vegetable and grain consumption rate,
8. Density of contaminated zone,
9. K_d of Np-227 in the contaminated zone,
10. Contaminated zone erosion rate,
11. Depth of soil mixing layer,
12. Density of unsaturated zone 1,
13. K_d of Pa-231 in unsaturated zone 3,
14. K_d of Th-229 in unsaturated zone 1,
15. K_d of Th-228 in the contaminated zone,
16. Plant transfer factor for Am,
17. K_d of Th-232 in the saturated zone, and
18. K_d of H-3 in unsaturated zone 1.

6.8.2.3.3 Calculation of Potential Dose from Discounted Radionuclides

Potential dose from discounted radionuclides was calculated using RESRAD v6.22 in the probabilistic mode using the decayed radionuclide concentrations provided in Table 6-4 and selecting the mean dose at specified times following license termination. The site-specific mathematical model developed in Section 6.6.2.1 was used with uncorrelated parameter distributions to perform the dose calculations. The RESRAD v6.22 parameters used were taken from Appendix 6-Y and the probabilistic parameter distributions and sensitive parameter value assignments were taken from Appendix 6-Z. Sensitive physical parameters were treated deterministically and non-sensitive physical parameters were treated stochastically. The dose factor library used for this calculation was the "RSNGS RF Dis Nuclide Dose" library containing the sensitive parameter values for plant transfer factors. The dose calculation times specified were 0, 1, 3, 25, 50, 75, 100, 300, 500 and 1000 years. The potential probabilistic mean dose at the specified calculated times from discounted radionuclides is provided in Table 6-19. A plot of the total dose (detected plus discounted radionuclides) versus years following license termination for the first 100 years following license termination is shown in Figure 6-14.

6.8.2.4 Calculated Total Dose for a Resident Farmer Scenario

As shown on Table 6-19 and Figure 6-14, the calculated total dose for a resident farmer scenario exceeds 25 mrem/y for approximately 30 years following license termination. However, it is highly unlikely that the District would consider public transfer of all or any portion of the impacted area of the site immediately or within 30 years upon completion of the FSS (partial site release) targeted as July 1, 2008 for the reasons given as justification for an industrial worker scenario in Section 6.4.2. Class B and C radioactive waste will be stored onsite in the Interim Onsite Storage Building (IOSB) under the existing 10 CFR Part 50 license for an indefinite period of time awaiting permanent disposal. Also, the Independent Spent Fuel Storage Installation (ISFSI) will be used to store spent fuel and greater than Class C radioactive waste under a 10 CFR Part 72 license for an indefinite period of time until transfer to the Department of Energy for permanent disposal. Furthermore, a new 500 MWe natural gas fueled cogeneration facility has been constructed on a non-impacted portion of the 2,480-acre site and began commercial operation in February 2006. Thirty years following completion of the partial site release is considered to be a reasonable time period during which the District would not consider public transfer of all or any portion of the impacted area of the site. Therefore, calculated dose for the resident farmer scenario for the one-year period starting July 1, 2038 is comparable to the calculated dose from the industrial worker scenario.

Table 6-19
Calculated Dose Using a Resident Farmer Scenario

Years Following License Termination	Detected Nuclide Dose (mrem/y)	Discounted Nuclide Potential Dose (mrem/y)	Total Dose (mrem/y)
0	7.76E+01	2.45E+00	8.01E+01
1	7.34E+01	2.33E+00	7.58E+01
25	2.90E+01	1.36E+00	3.04E+01
50	1.07E+01	9.50E-01	11.6E+01
75	4.45E+00	6.67E-01	5.12E+00
100	1.96E+00	4.59E-01	2.42E+00
300	9.00E-03	6.00E-03	1.50E-02
500	0.00E-03	4.00E-03	4.00E-03
1000	0.00E-03	3.00E-03	3.00E-03

6.8.3 Cattle Grazing Alternative Exposure Scenario

Portions of the non-impacted area of the 2,480-acre site are open range areas that are leased to local ranchers for cattle grazing. Typically, the open range areas produce an abundance of native grass during the winter and spring rainy season. After the rainy season is over the grass quits growing, dries out and becomes dormant. Portions of the impacted area of the site are also open range areas (e.g., the south storm drain outfall area and the liquid effluent pathway area). Historically, cattle grazing has occurred in the south storm drain outfall area; however, there are no provisions or expectations to preclude cattle grazing in any of the impacted range areas in the future. Therefore, the dose impact of maintaining an industrial worker scenario but allowing cattle grazing within impacted range areas outside the Industrial Area and consumption of meat from the grazing cattle by an offsite member of the public was evaluated.

Although characterization soil samples have shown this not to be the case, a basic assumption was made that all impacted open range areas are contaminated to the industrial worker scenario maximum allowable radionuclide soil concentrations from Table 6-6, similar to the assumption made in Section 6.8.1 to calculate dose from detected radionuclides under a resident farmer scenario.

To perform this evaluation, the calculation described in Section 6.8.2.2 was modified to create a cattle grazing scenario. These modifications included:

- Suppression of all pathways except meat ingestion,
- Turning off irrigation because the open range areas are not irrigated,
- Setting the “livestock water intake for meat” parameter to “0” because livestock drinking water is not provided from Rancho Seco wells, and

- Setting the "fraction of grain in beef cattle feed" parameter to "0" because no grain for cattle feed is grown on the Rancho Seco site.

With the above modifications, the calculated dose represented the maximum potential dose that a member of the public, whether residing onsite or offsite, could receive from consuming meat from cattle grazing on impacted open range areas that are uniformly contaminated to the industrial worker scenario maximum allowable radionuclide soil concentrations. The calculation showed that the maximum potential peak of the mean dose is 5.13 mrem/year.

Although a maximum potential peak of the mean dose of 5.13 mrem/year has been calculated for the cattle grazing alternative exposure scenario, this dose does not need to be accounted for in the industrial worker scenario dose limit because the offsite member of the public is distinctly different from the industrial worker and the two different scenarios do not impact each other.

Figure 6-1
Rancho Seco Reservoir and Recreation Area

Figure 6-2
Rancho Seco Switchyard and ISFSI

Figure 6-3
Rancho Seco Photovoltaic Generating Facility

Figure 6-4
Aerial Photograph of the Combined Cycle Cosumnes Power Plant

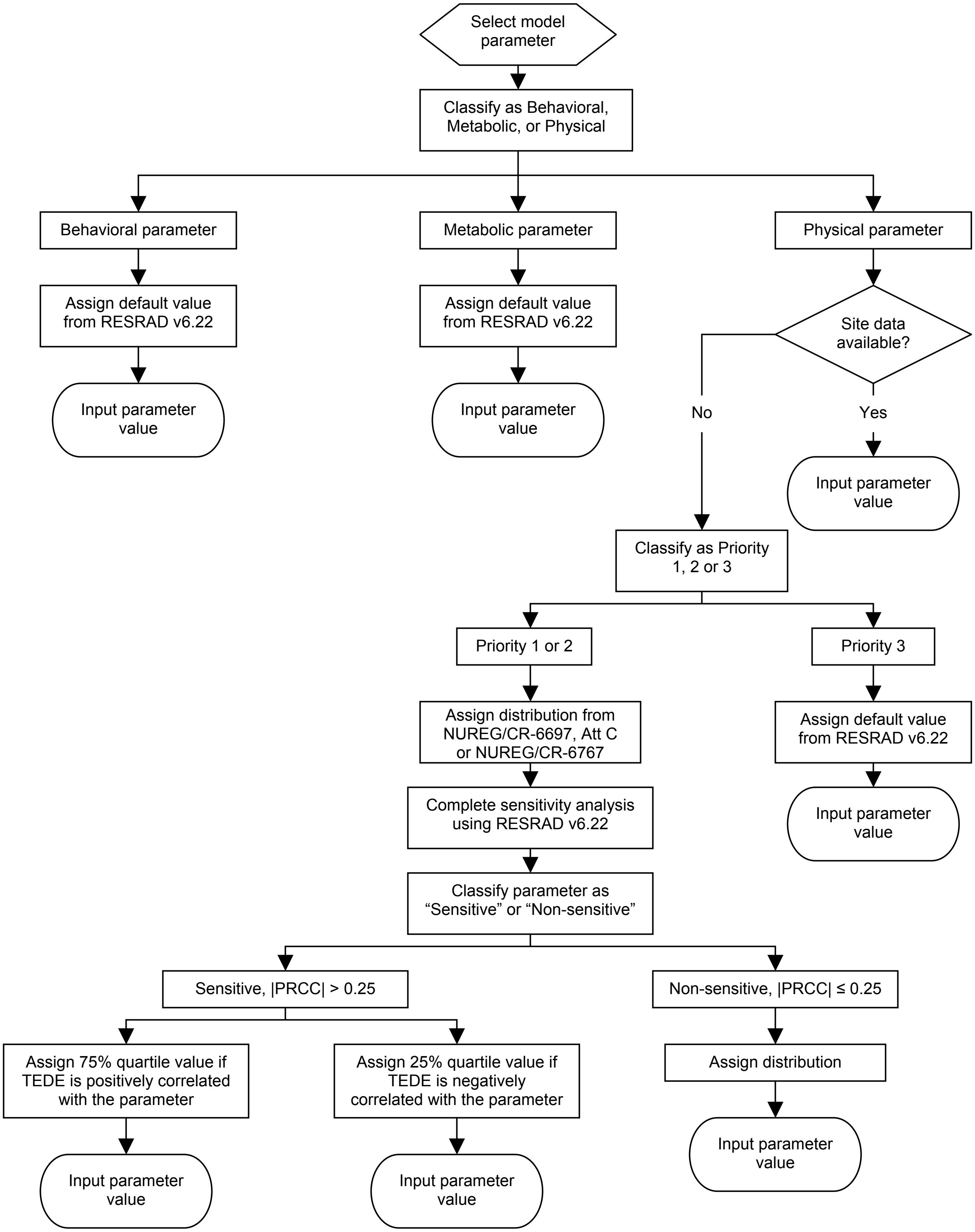

Figure 6-5
RESRAD Parameter Selection Process

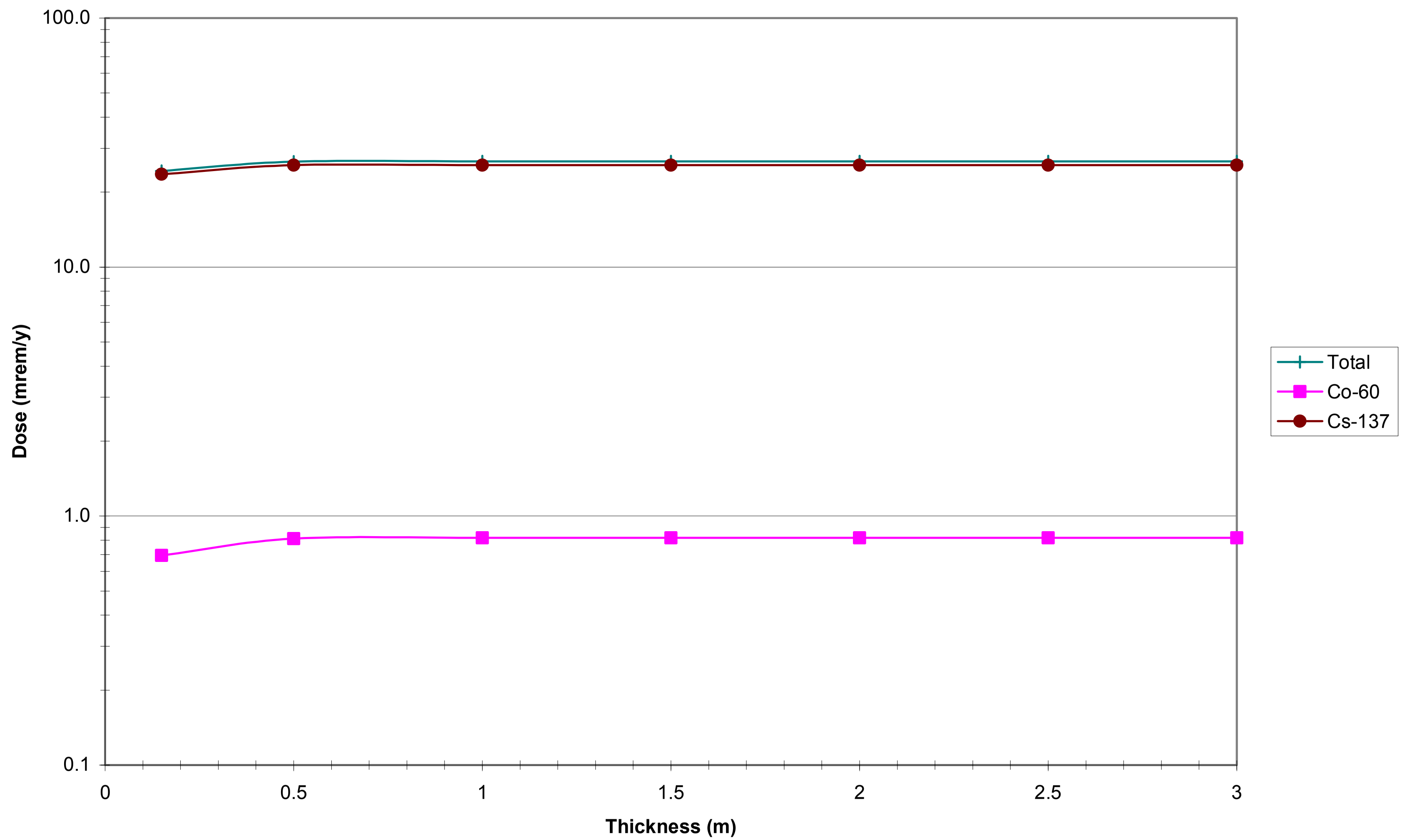

Figure 6-6
Peak of the Mean Dose vs Contaminated Layer Thickness for Principal Dose Contributors

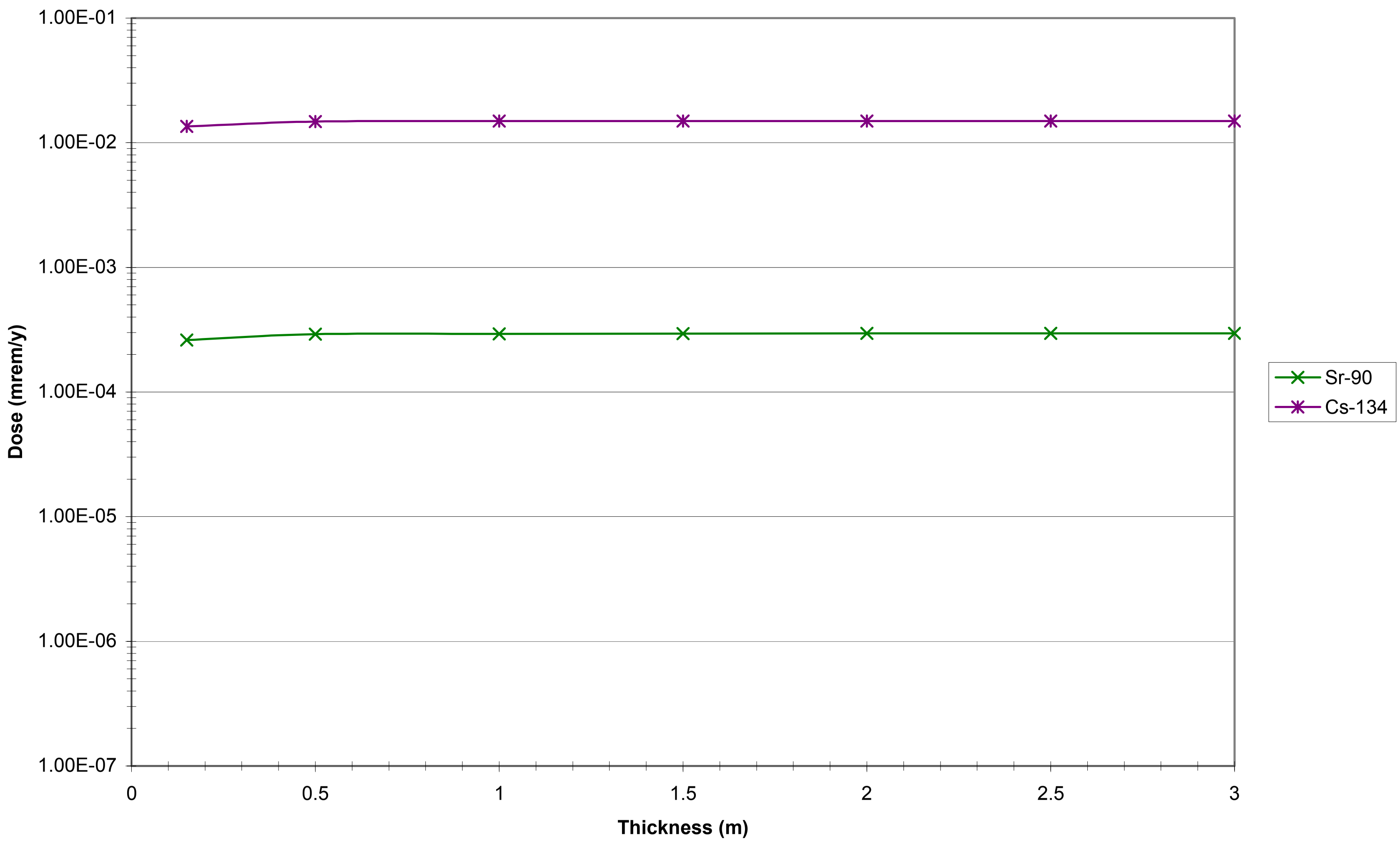

Figure 6-7
Peak of the Mean Dose vs Contaminated Layer Thickness for Minor Dose Contributors

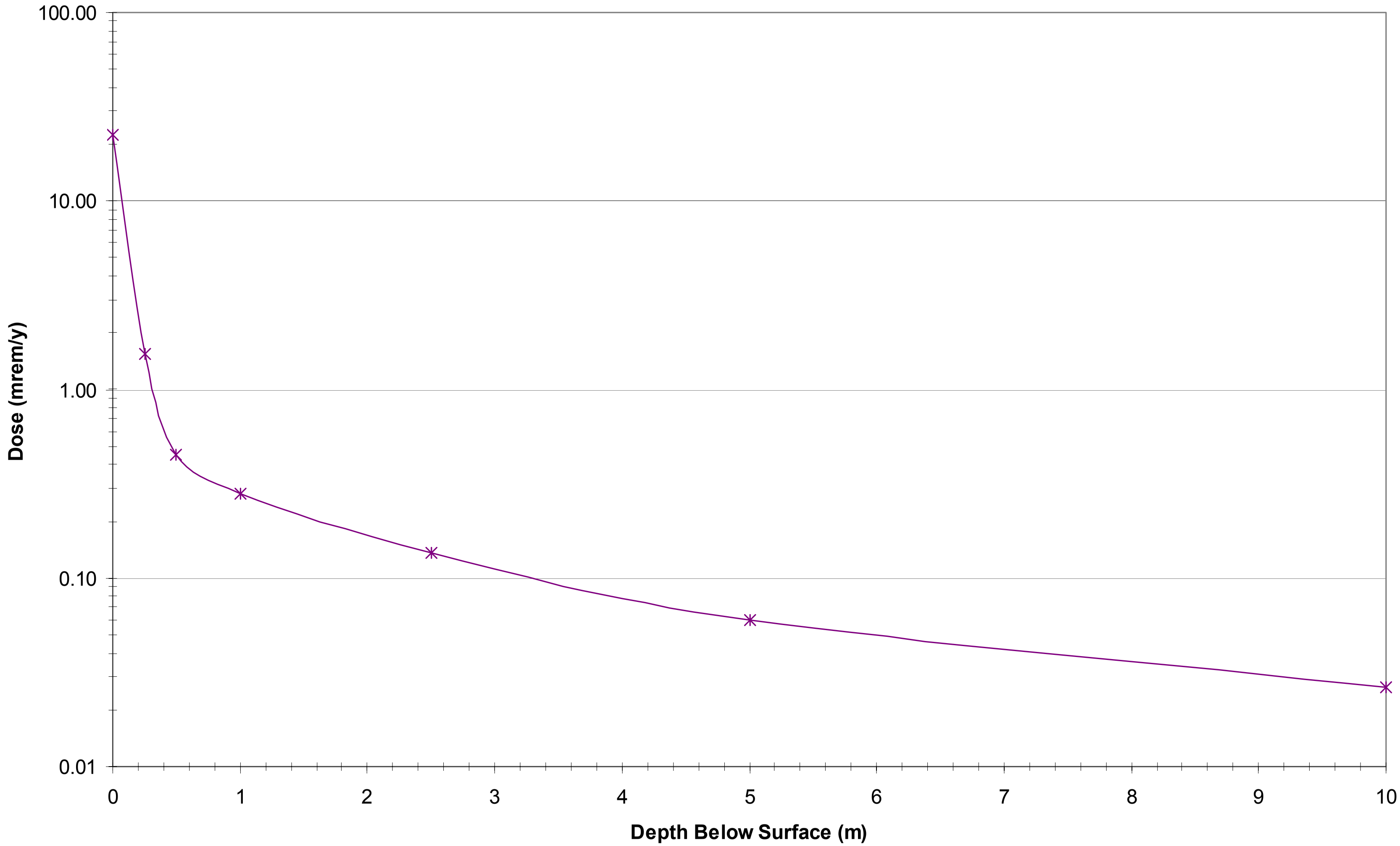

Figure 6-8
Peak of the Peak of the Mean Dose vs Discrete Contamination Pocket Depth

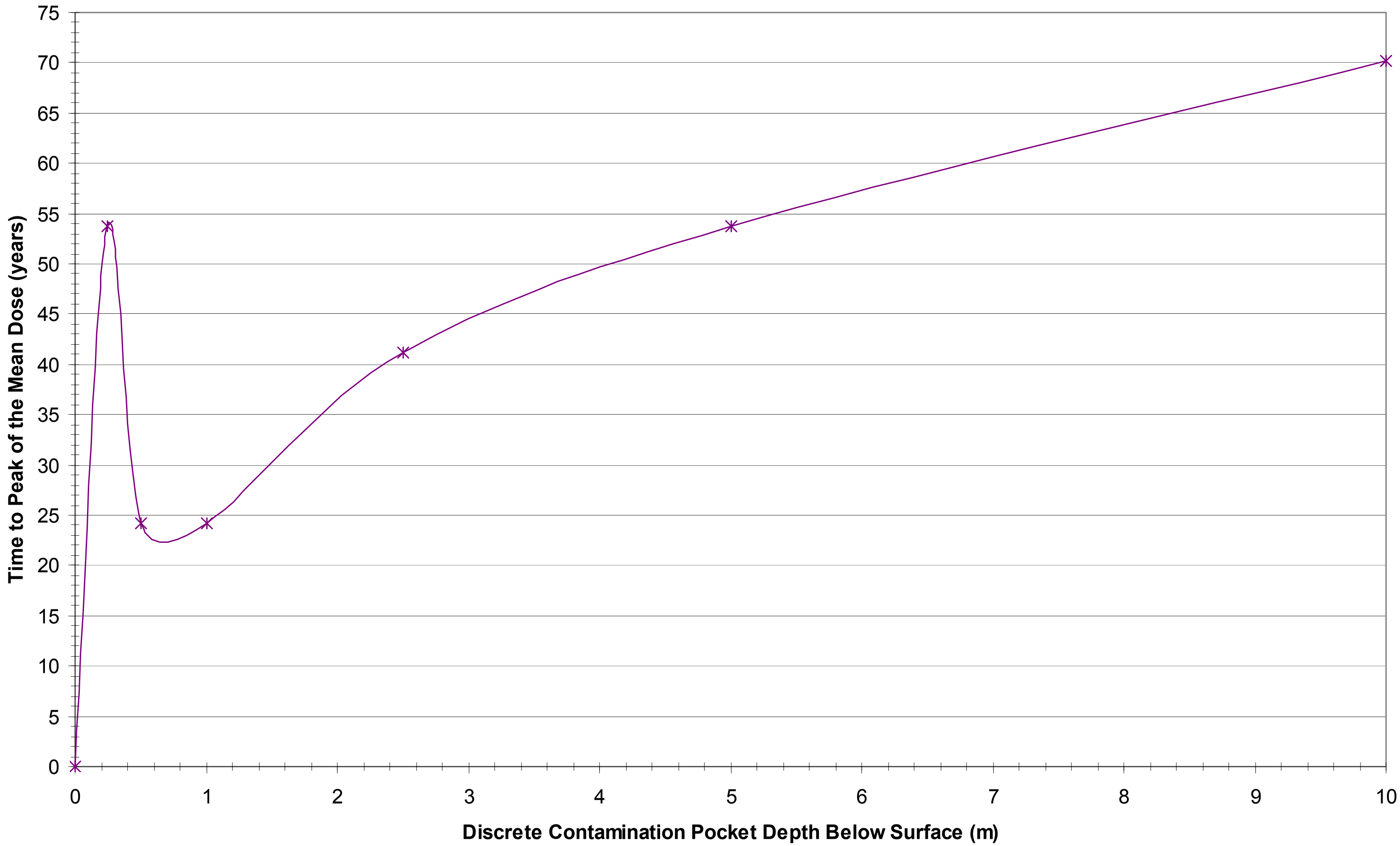

Figure 6-9
Time of **the Peak of the Mean Dose vs Discrete Contamination Pocket Depth**

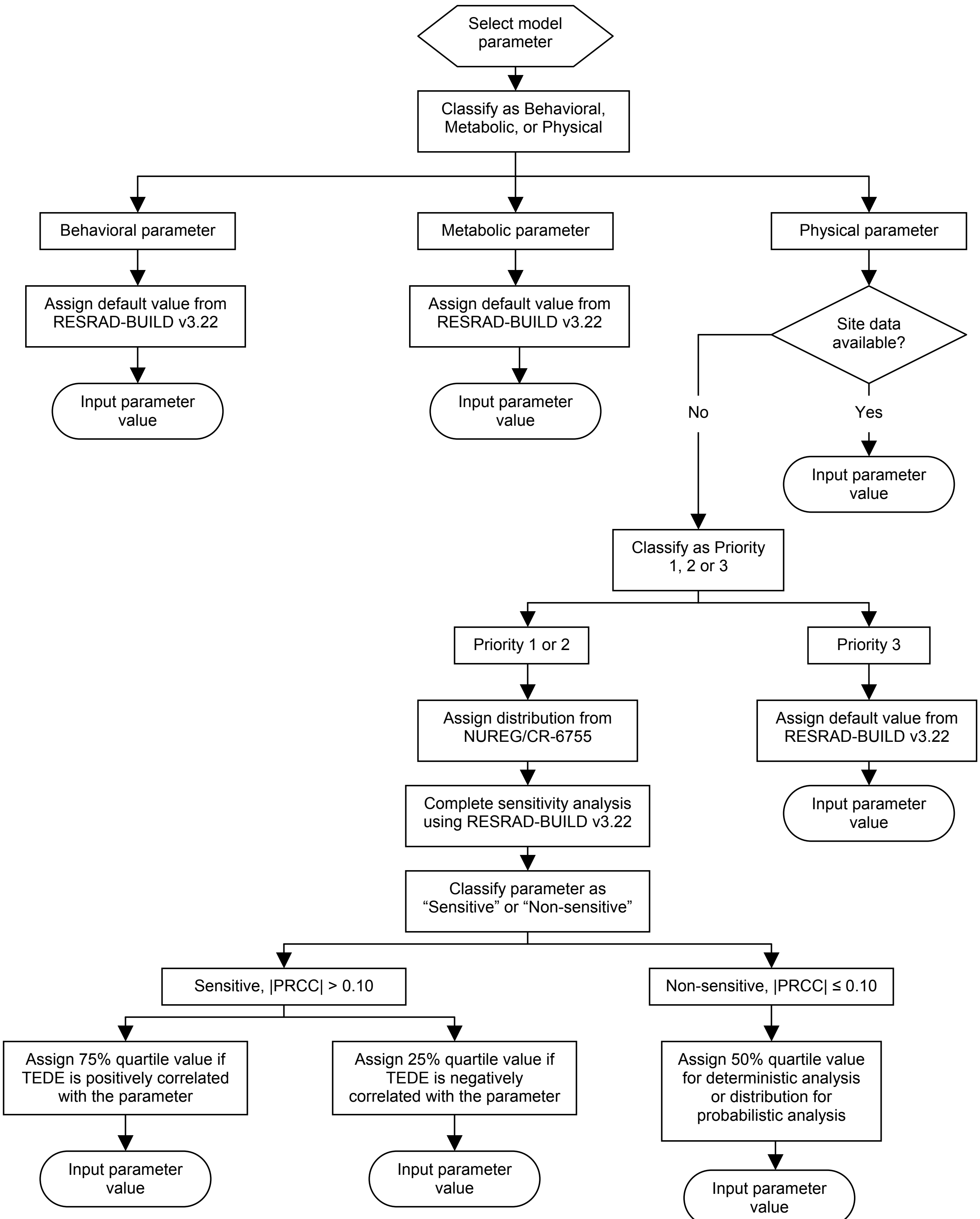

Figure 6-10
RESRAD-BUILD Parameter Selection Process

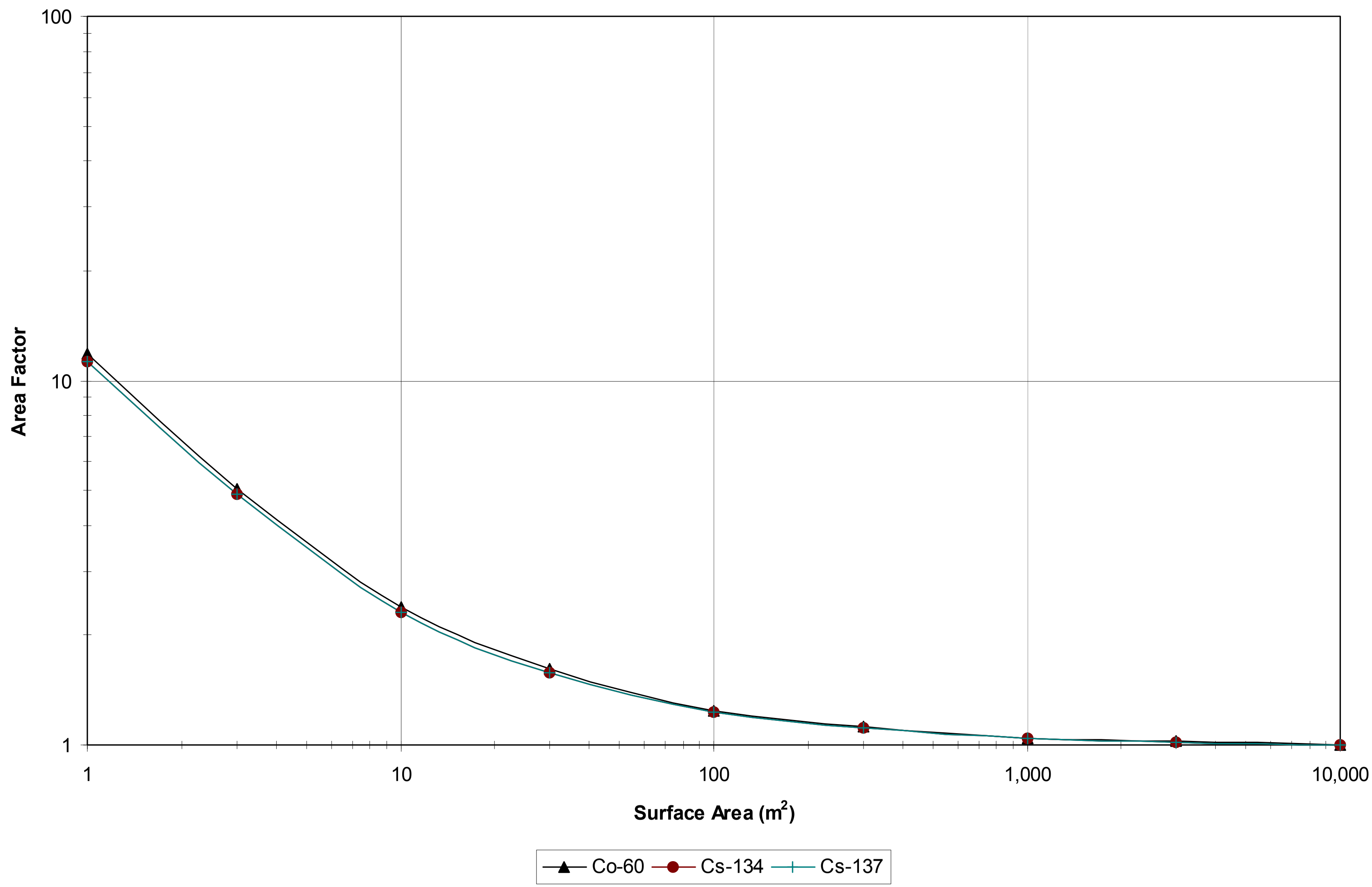

Figure 6-11
Surface Soil Area Factors for Gamma Emitters

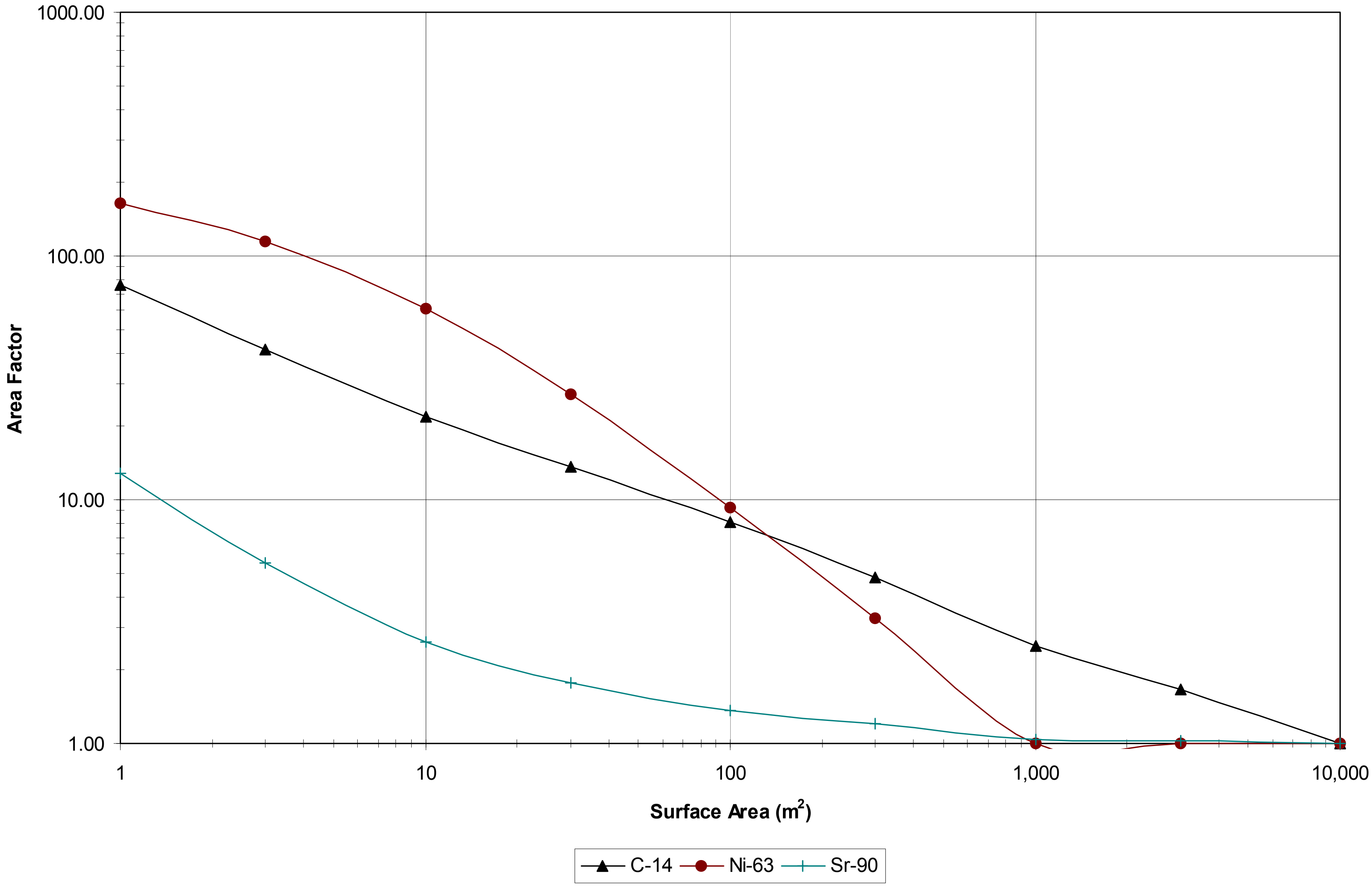

Figure 6-12
Surface Soil Area Factors for Beta Emitters

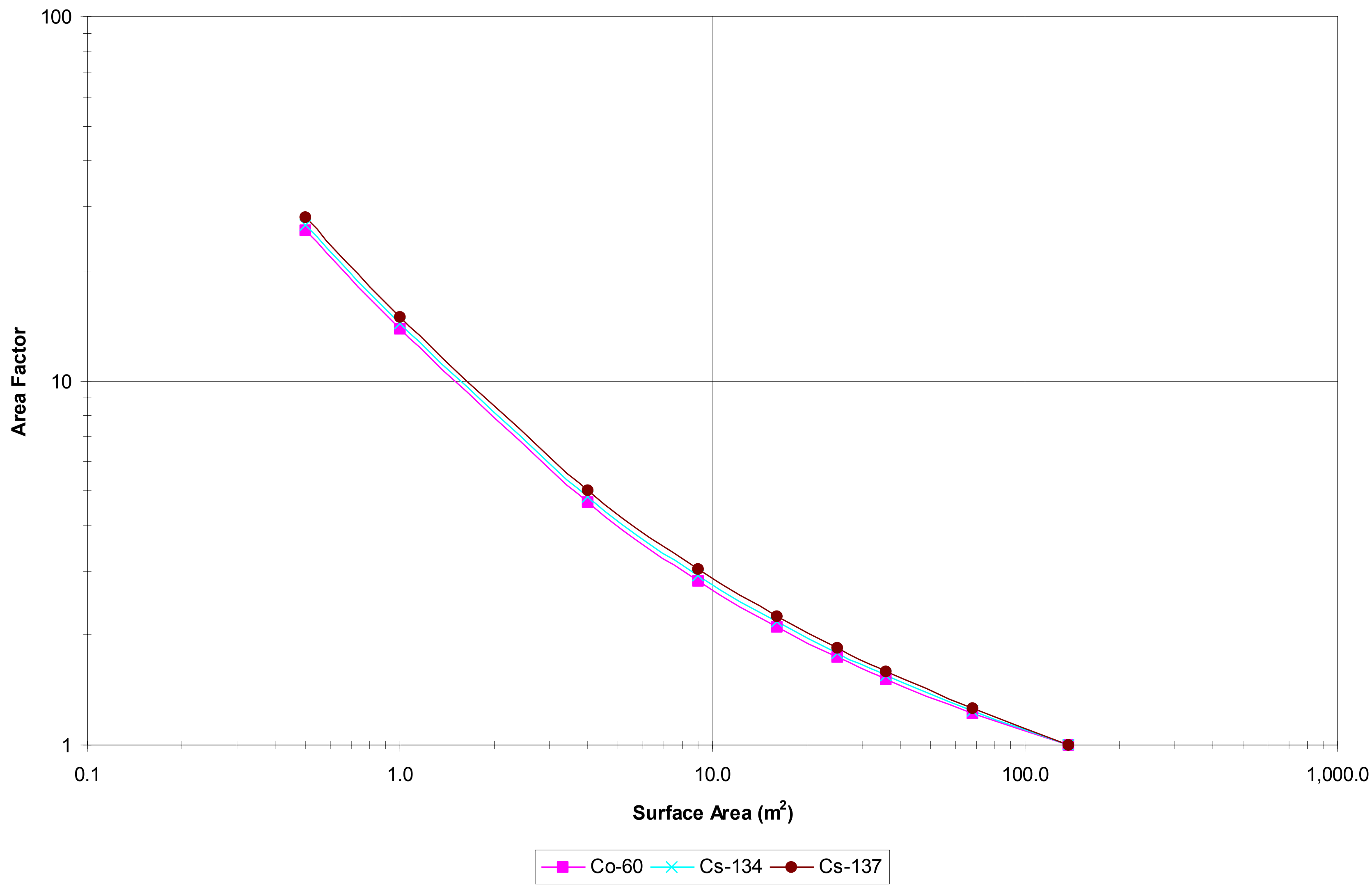

Figure 6-13
Structural Surface Area Factors

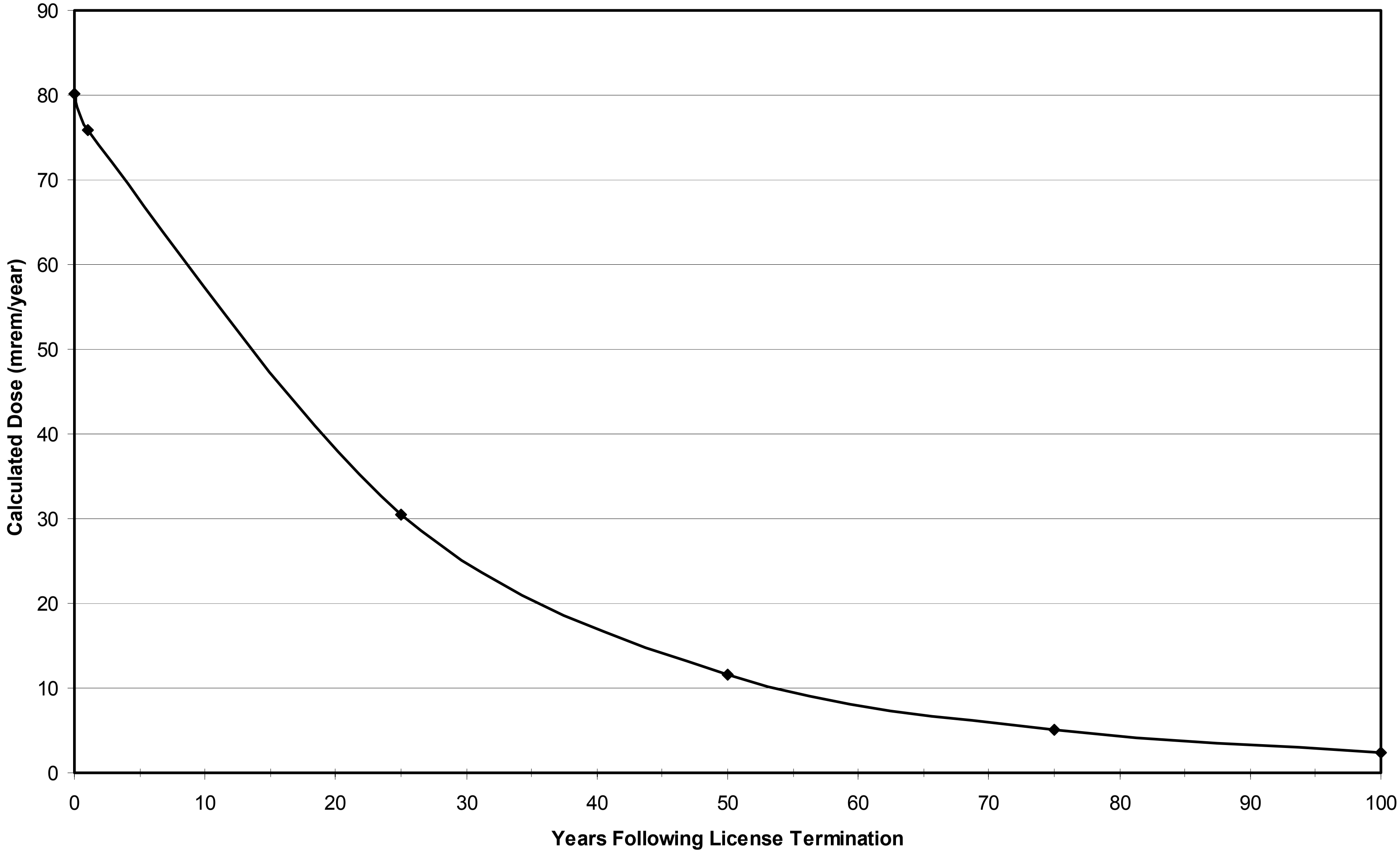

Figure 6-14
Calculated Dose Using a Resident Farmer Scenario

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